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# Assessment of ISLOCA Risk–Methodology and Application to a Babcock and Wilcox Nuclear Power Plant

Main Report

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Prepared for
U.S. Nuclear Regulatory Commission

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# ABSTRACT

This document presents information essential to understanding the risk associated with inter-system loss-of-coolant accidents (ISLOCAs). The methodology developed and presented in this document provides a state-of-the-art method for identifying and evaluating plant-specific hardware designs, human performance issues, and accident consequence factors relevant to the prediction of the ISLOCA risk. This ISLOCA methodology was developed and then applied to a Babcock and Wilcox (B&W) nuclear power plant. The results from this application are described in detail. For this particular B&W reference plant, the assessment indicated that the probability of a severe ISLOCA is approximately 2.2E−06/reactor-year.

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# EXECUTIVE SUMMARY

Inter-system loss-of-coolant accidents (ISLOCAs) are identified in some probabilistic risk assessments (PRAs) as major contributors to risk at nuclear power plants (NPPs). They can potentially result in core damage and containment bypass, which may lead to the early release of large quantities of fission products. Several operating nuclear reactors have experienced events that can be identified as ISLOCA related.

The occurrence of these operational events, loosely termed "ISLOCA precursors," have called into question the assumptions and boundary conditions typically used in PRAs. The scope of the questions concerned the ISLOCA frequency of occurrence, the type of potential initiators, and the means of identifying and mitigating this class of accidents. These questions and the operational experience suggest that the risk associated with an ISLOCA may be larger than previously estimated and that additional measures may be needed to prevent and/or control such accidents. Therefore, it is necessary to more fully understand the issues associated with ISLOCA sequences by formulating and implementing a U.S. Nuclear Regulatory Commission (NRC) sponsored ISLOCA Research Program.

The NRC's ISLOCA Research Program has several important objectives. These objectives cover the issues believed to dominate NPP risks for the ISLOCAs, and include providing qualitative and quantitative information on the hardware, human factors, and accident consequence issues. A risk assessment methodology was developed to support these objectives. This ISLOCA methodology has been applied to three pressurized water reactors: a Babcock and Wilcox (B&W) plant, a Westinghouse 4-loop plant, and a Combustion Engineering plant. This report describes the ISLOCA methodology and also presents detailed results from the application of the methodology to a B&W plant.

A methodology was developed to perform qualitative and quantitative evaluations for the ISLOCA sequences. The steps and their relationship are shown in Figure ES-1. The application of this methodology to the B&W plant was performed by a team consisting of both PRA and human factors specialists. The important results are displayed in Table ES-1. Insights and observations that are *specific to this reference-plant analysis* are as follows:

- Significant contributors to the ISLOCA core damage frequency (CDF) and risk were identified to be the result of human errors. These human errors were associated with routine plant operations [stroke testing high-pressure injection (HPI) valves] and during plant mode changes (shutdown). However, these types of errors have a much greater probability of being recovered before core damage can occur than do hardware failure initiated events.

- Although the ISLOCA scenarios that are influenced primarily by hardware failures are calculated to be significant contributors to CDF and the risk associated with an ISLOCA, this observation is not supported by the operational experience. Catastrophic failures of check valves *known to be initially seated, leak tight, and held closed by a large delta-pressure* are extremely rare (we could not find any reported instances). The CDF contribution of hardware failure initiated events is suspected to be the result of uncertainties in the data. Information on leak sizes for check valve failures in the available data is vague and incomplete.

- The isolation of the break is an important recovery action during an ISLOCA. The analysis indicates that hardware is typically available to isolate these ISLOCA ruptures. Procedures and training can be upgraded to improve the likelihood that this hardware will be used to isolate these breaks and thus reduce risk.

- A reduction in risk can be achieved by changes to procedures, training, and instrumentation. These changes can improve situational awareness (i.e., that personnel

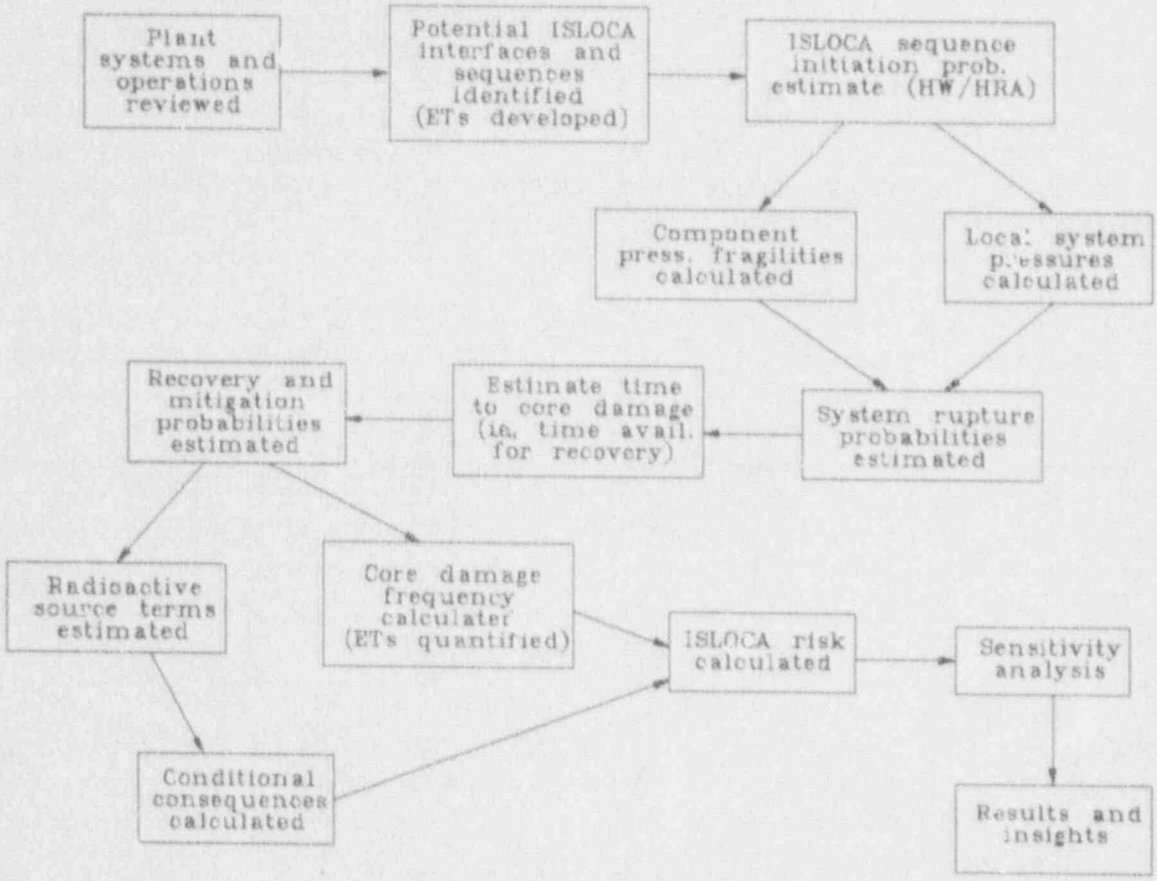


**Figure ES-1.** Approach for plant-specific evaluation of ISLOCA.

**Table ES-1.** Plant-damage state frequencies from ISLOCA sequences for a B&W reference-plant (frequency per reactor-year).

| Plant-damage state | Frequency | Plant-damage state description |
|---|---|---|
| OK-op | 1.1E−02 | Interfacing system is overpressurized, but does not rupture. |
| LK-ncd | 1.5E−03 | Reactor coolant is lost, but is either too small to be significant or is isolated before core damage occurs (no core damage). |
| LOCA-ic | 8.9E−08 | Sequence results in a loss-of coolant accident inside containment. |
| REL-mit | 0.0 | Core damage, but radioactive release is mitigated. |
| REL-lg | 2.2E−06 | Core damage with a large unmitigated radioactive release. |
| Total core damage frequency | 2.2E−06 | Sum of large and mitigated release frequencies. |

understand the consequences of procedural mistakes to ISLOCA sequences), and provide direction for the operators if an event were to occur.

- Environmental effects in the auxiliary building and their affect on vital equipment is an uncertain issue. Provided the equipment is environmentally qualified (e.g., based on a high-energy line break analysis), temperature and humidity should not be of concern. However, it is possible that large amounts of water will be released into the auxiliary building in an ISLOCA event, which could generate significant flooding in vital equipment areas. This aspect is influenced by several factors: the size of the rupture, drainage capacity (including drainage to other areas in the building), sump pump capacity, presence and capacity of fire suppression sprays, and the location of vital equipment relative to flooding sources. The consequence of this issue is the availability of equipment and time necessary for recovering from an ISLOCA. Once primary system makeup and heat removal capacity has been lost, there are few options left to the operators. The pertinent question is how much time do the operators have to isolate the rupture before all equipment is lost. For the B&W plant analysis, worst-case scenario estimates range from 2 to 4 hours for the large and small ruptures, respectively. (Note: These do not necessarily represent conservative estimates but simply the worst combination of several likely scenarios based on rupture size and location, and system and operator responses.)

Caution must be exercised in applying the results from the analysis to different B&W plants. Conclusions regarding the ISLOCA risk for the B&W plant analyzed in this study will likely be different for other B&W plants.

Because of the influence of potential human errors (both initiation events and post-rupture recovery events) on ISLOCA risk, evaluations for other plants should include a comprehensive human factors assessment. This assessment should include identifying possible human errors contributing to the initiation of an event (both latent and commission errors), and an assessment of operator performance in recovering from an ISLOCA rupture (i.e., detecting, diagnosing, and isolating ruptures). The human factors ISLOCA assessments must identify and evaluate the influences on human performance during all modes of operation.

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# ACRONYMS

| | |
|---|---|
| AFW | Auxiliary feedwater |
| B&W | Babcock & Wilcox |
| BWST | Borated water storage tank |
| CDF | Core damage frequency |
| CFT | Core flood tank |
| DF | Decontamination factor |
| DHR | Decay heat removal |
| DHR-SD | Decay heat removal-shutdown |
| DHR-SU | Decay heat removal-start up |
| ECA | Emergency contingency action |
| ECCS | Emergency core cooling systems |
| EF | Error factor |
| EPRI | Electric Power Research Institute |
| GLP | Gross leak pressure |
| HEP | Human error probability |
| HPI | High-pressure injection |
| HRA | Human reliability analysis |
| INEL | Idaho National Engineering Laboratory |
| IRRAS | Integrated Reliability and Risk Analysis System |
| ISLOCA | Inter-system loss-of-coolant accident |
| LER | Licensee Event Report |
| LOCA | Loss-of-coolant accident |
| LPI | Low-pressure injection |
| MAAP | Modular Accident Analysis Program |
| MACCS | MELCOR Accident Consequence Code System |
| MOV | Motor-operated valve |
| MU&P | Makeup and purification |
| NPP | Nuclear power plant |
| NRC | U.S. Nuclear Regulatory Commission |
| P&ID | Piping and instrumentation diagram |
| PIB | Pressure isolation boundary |
| PORV | Power-operated relief valve |
| PRA | Probabilistic risk assessment |
| RCS | Reactor coolant system |
| RO | Reactor operator |
| RPV | Reactor pressure vessel |
| SDC | Shutdown cooling |
| SHARP | Systematic Human Action Reliability Procedure |
| SG | Steam generators |
| TALENT | Task Analysis-Linked Evaluation Technique |
| THERP | Technique for human error rate prediction |

# Assessment of ISLOCA Risks—Methodology and Application to a Babcock and Wilcox Nuclear Power Plant

## 1. INTRODUCTION

The *Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400,[1] identified a class of accidents that can result in overpressurization and rupture of systems that interface with the reactor coolant system. These events were postulated to be caused by the failure of the check valves and motor-operated valves (MOVs) normally used for system isolation. A subset of these inter-system loss-of-coolant accidents (ISLOCAs) were called V-sequences or event V. These sequences were characterized by the failure of MOVs and/or check valves, and the rupture of low-pressure piping outside of the containment building. Some event-V ISLOCAs were shown to be significant contributors to risk because the rupture caused core damage, and fission products bypassed the containment and were discharged directly to the environment. Subsequent probabilistic risk assessments (PRAs), including NUREG-1150[2] results for Surry and Sequoyah, have identified ISLOCAs as important contributors to public health risk. Researchers at Brookhaven National Laboratory have evaluated the vulnerability of reactor designs to an ISLOCA and identified improvements that would reduce ISLOCA initiation frequency.[3,4]

Recent events at several operating reactors have been identified as precursors to an ISLOCA. These events have raised questions about the previously assumed frequency of occurrence, potential initiators, and means of identifying and mitigating this potential accident; suggesting that the risk associated with an ISLOCA may be larger than previously estimated and that additional measures may be needed to prevent and/or control these accidents. In response to these questions, a June 7, 1989, memorandum, "Request for Office of Nuclear Regulatory Research (RES) Support for Resolution of the ISLOCA Issue," was transmitted from Dr. Thomas E. Murley to Mr. Eric S. Beckjord. The ISLOCA Research Program described in this report was initiated as a result of this memorandum.

The objective of the ISLOCA Research Program is to provide the U.S. Nuclear Regulatory Commission (NRC) with qualitative and quantitative information on the hardware, human factors, and accident consequence issues that dominate the nuclear power plant (NPP) risk associated with an ISLOCA. To accomplish this objective, a methodology has been developed that is based on PRA, human factors, and human reliability analysis (HRA) techniques. This methodology can be used for the following:

- Identify the risk contribution from both hardware failures and human errors issues and to develop recommendations for risk reduction.
- Identify the effects of specific types of human errors and their root causes, on ISLOCA risk.
- Evaluate the fragility of low-pressure systems exposed to high-pressure, high-temperature reactor coolant. These evaluations include identification of likely failure locations and estimates of probabilities of failure.
- Identify and describe potential ISLOCA sequences including sequence timing, possible accident management strategies, and the effects of possible ISLOCAs on other equipment and systems.
- Estimate the consequences associated with postulated ISLOCA events, including estimates of source terms and offsite consequences. Again, important issues can be

identified and recommendations can be made on possible consequence reduction actions.

The methodology developed to estimate the core damage frequency (CDF) and risk associated with an ISLOCA has been applied to three pressurized water reactors: a Babcock and Wilcox (B&W) plant, a Westinghouse 4-loop ice condenser plant, and a Combustion Engineering plant. This report describes the ISLOCA methodology and documents its application to the B&W plant.

Section 2 of this report describes the methodology developed to evaluate the effects of an ISLOCA. Section 3 contains a description of the interfacing systems and the possible ISLOCA sequences for the B&W plant being examined. Section 4 describes the plant-specific results from the assessment of ISLOCA at the B&W plant, and Section 5 contains the observations and insights based on this assessment. Appendices A through M of this report are used to document the details of many of the evaluations.

# 2. APPROACH

A methodology that is based on PRA, human factors, and human reliability analysis techniques has been developed for estimating the risk associated with an ISLOCA at an NPP. The steps in this individual plant methodology are illustrated in Figure 1. Subsections 2.1 through 2.8 briefly discuss each of the steps.

The first step in the development of this methodology was a review of historical plant operating information publicly available in the United States. This review included an identification and evaluation of all of the Licensee Event Reports (LERs) that (a) involved valve failures resulting from either hardware or human causes or (b) indicated an ISLOCA had occurred. The results of this review provided information on the causes and frequencies of valve failures and provided important insights on the systems involved and the potential causes of ISLOCAs that have occurred. This information was used to identify systems to be reviewed during the development of the events in the event trees, and for quantification of the failure rates of some interfacing system valves. Appendix A provides a brief summary of the historical experience related to ISLOCA events.

## 2.1 Assess Potential For ISLOCA

The initial step in the individual plant evaluation approach is to make a preliminary assessment of the potential for an ISLOCA. Hardware and operating information on a wide range of low- and high-pressure interfacing systems must be collected. This required information includes

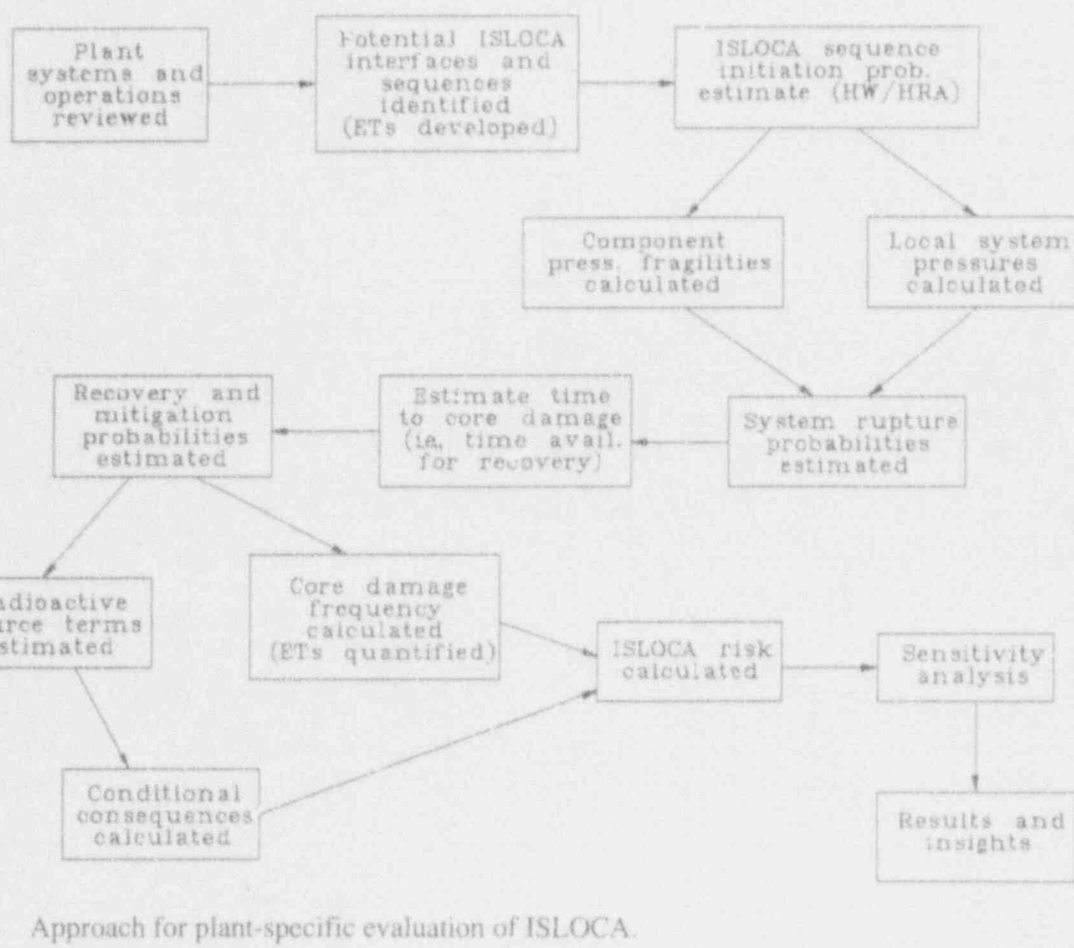


**Figure 1.** Approach for plant-specific evaluation of ISLOCA.

plant procedures, piping and instrumentation diagrams (P&IDs), isometric drawings, training manuals, etc. This information must be reviewed by the team of PRA and human factors specialists involved in the analysis. This review will allow them to become familiar with the systems and operations that have the potential to initiate, prevent, or mitigate an ISLOCA. All systems that interface with the reactor coolant system (RCS) must be identified. The maximum interfacing system break size that will not result in core damage must be determined. The interfacing systems are then screened and categorized in terms of this pipe size and the potential for containment bypass. The interfacing systems are screened based on (a) systems with pipe sizes larger than a specified maximum and (b) those systems that could bypass the containment. The systems that meet the screening criteria are analyzed further to identify potential ISLOCA initiators and sequences. The identified sequences are developed in sufficient detail to guide a team of PRA and human factors specialists in obtaining detailed information during an extended plant visit.

## 2.2 Gather Detailed Plant-Specific Information

An extended visit to the plant is necessary to gather the information needed to complete the review, development, and assessment of the candidate ISLOCA sequences. Members of the team that developed the candidate sequences will gather the needed information by interviewing operations personnel and analyzing the systems of interest. The types of information that are obtained during an extended visit include detailed information on

- Hardware that could be involved in an ISLOCA (i.e., control valves, relief valves, piping, flanges, pumps, and heat exchangers).
- Procedures and guidelines imposed on plant personnel during startup, normal power operation, and shutdown of the plant. Also, detailed information on maintenance and in-service test practices is required.
- Factors that could influence plant personnel performance as it relates to initiation, detection, prevention, or mitigation of an ISLOCA.

## 2.3 Develop Event Trees

After specific plant information is collected, a final list of interfaces and sequences is generated and the detailed analysis begins. This analysis is performed through a joint effort of the PRA and human factors specialists. The sequences are modeled using component level event trees that combine the hardware faults and the human errors that compose each sequence. Generally the event trees comprise three phases:

1. The initiating events, which are those combinations of failures, both hardware and human related, that result in a breach of the pressure isolation boundary (PIS) and allow high-pressure RCS water to enter the lower pressure interfacing system
2. The rupture events that identify the probability of a rupture in the interfacing system (size and location)
3. The post-rupture events that identify the actions and estimate the success of the control room operators in recovering from an ISLOCA or in mitigating its consequences.

## 2.4 Estimate Rupture Potential

The performance of plant components that are designed for low-pressure conditions and are exposed to the high pressures associated with an ISLOCA must be assessed. The methodology for performing this assessment is as follows:

- An event tree model of each system is built that will compare the estimated failure pressure with the expected local system pressure for each important component. This model is constructed and input to the EVNTRE computer code,[5] which was developed for the NUREG-1150[2] program, for the assessment of complex event trees.

- The failure probability of each piece of equipment in the low-pressure rated system is described as a lognormal distribution with a specified median failure pressure and standard deviation.

- The thermal-hydraulic behavior of the various systems is calculated to estimate the pressure distributions in the system based on (a) the expected initiating event, (b) the initial primary system conditions, and (c) the expected performance of relief valves designed to protect the systems.

- Each question in the event tree is answered by (a) randomly selecting a failure pressure from the failure pressure distribution of the appropriate component, and (b) comparing the selected component failure pressure with a selected system pressure. The system pressure is randomly selected based on the expected operating conditions and assuming a normal distribution with an estimated mean and standard deviation. If the sampled component failure pressure is below the sampled system pressure, the component is assumed to have failed. Otherwise no failure is assumed. Each component in the low-pressure rated system is evaluated in this manner until all questions in the event tree have been examined. This process is repeated approximately 10,000 times in a Monte Carlo simulation. This sampling is feasible because of the relatively small size of the EVNTRE model.[5]

- Once the simulation is completed, the output is binned and the relative frequency of various equipment failures can be estimated (if system overpressurization exists).

The component and piping failure pressures used for the B&W reference-plant rupture calculations were developed using an independent structural analysis. This analysis was performed by Wesley et al.[6] Not only were failure pressures calculated, but expected leak rates and leak areas were calculated as well. In this respect, flanges exhibit a somewhat unique behavior in that there are actually three failure pressures of interest. First, is the estimated gross leak pressure (GLP), at which a measurable leak area appears. At lower pressures (starting at some fraction of GLP, $P_x$), seepage around a gasket is possible but at very small rates (measured in mg/sec). Once the GLP is exceeded, the bolts in the flange begin to stretch (elastically) and the flange surfaces begin to separate. At higher pressures ($P_0$), the bolts begin to yield plastically. At this point, large leak areas begin to appear with corresponding large leak rates. These three pressure regions (between $P_x$ and GLP, between GLP and $P_0$, and greater than $P_0$) are associated with three sizes of leaks: spray leaks, small leaks, and large leaks.

## 2.5 Perform Human Reliability Analysis

HRA was used to model the important human errors for each scenario in the B&W ISLOCA PRA. HRA is a methodological tool that involves the quantitative analysis, prediction, and evaluation of work-oriented human performance. HRA can be used to determine which factors in the system lead to less-than-optimal human performance. As a diagnostic tool, HRA can estimate the error rate anticipated for individual tasks and can determine where errors are likely to be most frequent.

The general methodological framework for the ISLOCA HRA was devised using guidelines (under development) from the NRC-sponsored Task Analysis-Linked Evaluation Technique (TALENT) Program,[7] which recommends task analyses, time-line analyses, and interface analyses as appropriate techniques for use in a detailed HRA. NUREG/CR-1278, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications* [which discusses the techniques for human error rate prediction (THERP)],[8] recommends similar techniques and provides a data base that can be used to generate human error probabilities (HEPs). Finally, the ISLOCA HRA integrated the steps from the *Systematic Human Action Reliability Procedure* (SHARP),[9] and *A Guide for General Principles* *of Human Action Reliability Analysis for Nuclear Power Generation Stations* (the draft IEEE Standard P1082/D7).[10]

This combination of approaches resulted in 11 basic steps to be followed in performing the HRA. These eleven steps include the following:

1. Select and train the team on plant functions and systems (IEEE P1082).[10]

2. Familiarize the team with the plant (IEEE P1082).[10]

3. Ensure that the many possible types of human actions and interactions are considered in the analysis (SHARP, IEEE P1082).[9,10]

4. Build the initial plant model (model systems and interactions) (IEEE P1082).[10]

5. Identify and screen specific human actions that are significant contributors to the safety and operation of the plant. This was accomplished through detailed task analyses, timeline analyses, observations of operator performance, and evaluations of the human/machine interface (SHARP and IEEE P1082).[9,10]

6. Develop a *detailed* description of the important human interactions and associated key factors necessary to make the plant model complete. This should include the key failure modes, identification of errors of omission/commission, and review of relevant performance shaping factors (SHARP) (IEEE P1082).[9,10]

7. Select and apply the appropriate HRA techniques for modeling the important human actions (SHARP).[9]

8. Evaluate the impact of significant human actions identified in Step 6 (SHARP).[9]

9. Calculate probabilities for the various human actions and interactions, determine sensitivities, and establish uncertainty ranges (SHARP and IEEE P1082).[9,10]

10. Review results for completeness and relevance (IEEE P1082).[10]

11. Document all information necessary to provide an audit trail and to make the information understandable (SHARP).[9]

Because most of the human actions in this HRA involved the use of various normal, abnormal, and emergency operating procedures, THERP-type HRA event trees[8] were used to model most of the human actions in the detailed analysis. However, not all ISLOCA scenarios were best represented by these THERP event trees alone. In those cases, HRA fault trees were used in conjunction with the typical THERP event trees. Detailed analyses were conducted using the fault trees and THERP event trees to estimate the probability of human error for each of the dominant human actions.

These event trees traditionally model human performance through the use of a diagram like that shown in Figure 2, with operator error generally placed along the descending right branches of the event tree and successful operator actions sequenced on the left side of the tree. For example, on the top left, event a (operators select RCS small leak procedure BW-OP-2522) is the success path. Failure to accomplish this task is modeled as event A (operators fails to select small leak procedure). When a second operator is involved, such as in event B (second operator fails to select small leak procedure), the action of this second operator may be modeled in a recovery branch, as shown in Figure 2. Since the second operator is in the control room in this scenario, the operator also has an opportunity to select BW-OP-2522, the small leak procedure. If successful, this becomes a recovery action because it would bring the model back to the success path (via the dotted lines in Figure 2).

Individual error branches on each of the HRA event trees (see Appendix E for details) were quantified using techniques from THERP, NUCLARR,[11] and engineering judgement.

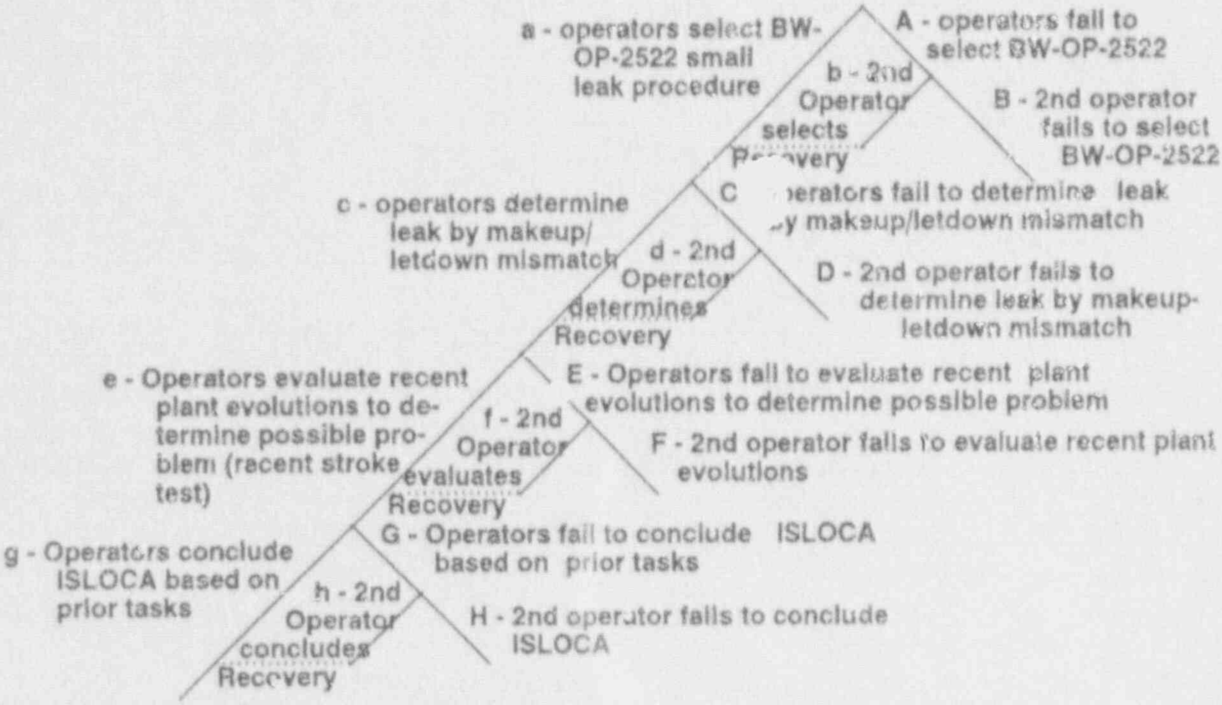


**Figure 2.** HRA event tree for HDA2-MU (operators fail to diagnose ISLOCA).

Specific human actions were assigned an estimate of a basic or unmodified HEP. These basic HEP estimates were then revised using performance shaping factors (PSFs) to realistically describe the work process at the plant. Each PSF was either positive or negative and accordingly, either decreased or increased the likelihood of a given human error. For example, an analog meter, such as a pressure gauge, which does not have easily seen limit marks, may be judged to have a negative PSF, and there would be a higher probability for human error in reading the gauge. Individual PSFs were derived from the task analyses, timeline analyses, evaluation of the human/machine interface, and direct observations of operator performance. They are presented as part of the *ISLOCA Inspection Report*.[12]

Specific PSFs that were investigated include

- The quality of the human/machine interface
- Written procedures (emergency, abnormal, maintenance, etc.)
- P&IDs
- Response times for systems and personnel
- Communication requirements
- Determination of whether the operator actions were skill, rule, or knowledge-based
- Crew experience
- Levels of operator stress in different scenarios
- Feedback from the systems in the plant
- Task dependence and operator dependence
- Location of the task (control room, auxiliary building, etc.)
- Training for individual operator actions including those required for ISLOCA situations.

Finally, all possible failure paths (i.e., sequences that included either single or multiple human errors leading to a failure of the action modeled by the HRA tree) were identified and used to estimate the total failure probability for the action modeled in the HRA tree, in accordance with the THERP guidelines. As depicted by Figure 2, each human error event tree may have several unique error paths. For example, event A and event B constitute an error path in which the first reactor operator (RO) fails to select BW-OP-2522, the small leak procedure (event A). This error action is followed by the failure of a second RO to select the same procedure (event B). In a similar manner, failure path A–b–C–D models a sequence where the RO fails to select the small leak procedure, the second RO recovers from this error by correctly selecting BW-OP-2522 (event b), only to have both ROs fail at actions C and D, the steps that would determine if there was a leak by comparing the rate of makeup to the rate of letdown. Probabilities for each unique error path were calculated by multiplying each HEP on a given error path by other HEPs on the same path. For example, the error rate for path A–B would be calculated by multiplying the HEP of failure A by that for failure B, resulting in a nominal HEP for that specific path. Other error paths for this event tree include A–b–c–E–F, a–c–E–F, and a–C–D, etc. The individual error path failure probabilities were then summed to give the total event tree failure probability. Comprehensive details of this process are provided in Appendix D and Appendix E for each event, and the results are summarized in Section 4.2.

Comparisons were made between the ISLOCA HRA methodology and other recent PRA/HRA efforts. Specifically, the HRA for Sequoyah Unit-1,[13] one of the five plants selected as part of the NUREG-1150 effort,[2] was used. Although Sequoyah is a Westinghouse plant and not a B&W plant, the HRA techniques are independent of the plant type. Sequoyah was chosen because it is typical of the latest generation of NRC-sponsored PRAs. The following observations are based on this comparison:

- Both the B&W ISLOCA and Sequoyah HRA analyses identified latent human errors.

- Both sets of analyses made use of quantitative techniques and failure rate probabilities supplied by NUREG/CR-1278.[8]

- The B&W HRA for ISLOCA makes use of a "multimethod" approach to HRA, incorporating HCR quantification techniques, as well as NUCLARR and THERP values for deriving HEP estimates. The Sequoyah analysis made use of only one method.

- The B&W HRA for ISLOCA suggests that HRA (or THERP type) event trees can be used in conjunction with HRA fault trees to develop accurate models representing sequences of human error actions.

- Both HRA analyses accounted for anticipated stress levels in everyday work environments.

- Both the Sequoyah and the ISLOCA HRAs used rates that were reflective of the adequacy of procedures associated with key maintenance and operations actions. Both studies identified human actions at the fault and event tree level. Section 4.8-2 of the Sequoyah PRA notes that "all errors identified were errors of omission."[13] Errors identified in the HRA for ISLOCA went beyond this and quantified various errors of commission (see Appendix E).

- Both sets of analyses postulated human action scenarios involving valve restoration after maintenance or inservice test. The Sequoyah analysis looked at miscalibration errors as contributing to core melt probabilities. This type of error did not figure as prominently for the B&W ISLOCA scenarios where improper valve lineups dominated.

- Valve misposition errors were of omission type only in the Sequoyah analyses, both omission and commission errors were accounted for in the B&W HRA.

- In the Sequoyah analysis, errors were assessed to be insignificant if valve position was annunciated in the cont[illegible] room. For one ISLOCA scenario, there is no annunciation available for HP-27 and HP-29 (local-manually operated valves).

- For analysis of post initiator operator actions, the Sequoyah analysis made use of emergency contingency action (ECA) procedures for Westinghouse plants. Furthermore, as part of the ground rules, it was assumed that operators would read each step in the appropriate procedure and then perform that step properly. At the reference B&W plant, there are no ISLOCA procedures available, and it is assumed, until data are discovered to the contrary, that there is an inherent background error rate in the reading and execution of procedures.

- The B&W ISLOCA HRA incorporated PSFs into the quantification of HEPs and did an uncertainty analysis for each HEP using Integrated Reliability and Risk Analysis System (IRRAS).[14] In addition, the B&W HRA for ISLOCA presents a sensitivity analysis for the case where performance shaping factors were optimized.

The findings clearly show that there were similarities and differences between the ISLOCA and Sequoyah HRA. Although there were similarities, such as the use of NUREG/CR-1278 quantitative techniques and stress levels,[8] the differences between the analyses suggest that the B&W ISLOCA HRA is more comprehensive than the Sequoyah HRA and extends current HRA efforts in providing HEP estimates for certain errors of commission.

## 2.6 Quantify Event Trees

After the likely sequences are identified and corresponding event trees developed, each individual event in the event tree is quantified. These event quantifications took many forms, depending on the event itself. In some instances, a single failure rate estimate is most appropriate (e.g., for a single valve failure event). In others a very complicated model development with Monte Carlo simulation (as in the case of the rupture probabilities) is required. The event tree is the mechanism by which all the individual calculations and estimates are integrated into the ISLOCA model. Upon generating an estimate for each of the individual events in the event tree (including accounting for dependencies and conditional events), the ISLOCA sequences and scenarios are quantified. For this particular phase of the B&W reference-plant analysis, the ETA-II PC-based computer program was used.[15]

## 2.7 Consequences

The formulation of the consequence methodology was guided by the objective to identify issues and concerns generic to the entire industry. This objective precluded the performance of a site-specific ISLOCA consequence analysis for the reference B&W unit. An appropriate consequence assessment therefore was based on the demographics of an average plant site and a source term from a "generic" B&W unit.

The demographics of the plant site are needed to provide an estimate of the consequences of an ISLOCA. The demographic information used in the consequence assessment of the ISLOCA methodology is selected to provide a normalized basis for estimating the risk at a number of different plants. The details on how this is accomplished are described in Appendix I. In general, the normalized consequences basis is developed by forming a nation-w[illegible] average site. This average site was develope[illegible] using the Sandia Siting Study[16] and the wea[illegible] weighted population densities for all the doc[illegible]ented sites. The average population density was then compared to the five sites studied in the NUREG-1150 program.[2] Ultimately, the Surry plant site was chosen because it most closely matched the average population density. The normalization of risk to an average site's population is an approach that should facilitate the drawing of generic conclusions.

A generic source term was also formulated for the ISLOCA B&W analysis. The source term was developed by comparing the B&W unit being analyzed in this study to other B&W units that have reported ISLOCA source terms. As a result of these comparisons, the ISLOCA source term profile (e.g., radionuclide release fractions) from the Oconee plant were utilized.[17] The source term was then developed using these release fractions and the radionuclide inventory of the Oconee plant scaled to the reference plant of this study. The radionuclide inventory scaling was accomplished by a multiple of the ratio of the powers of the two different units.

The source term that was developed for the Oconee ISLOCA sequence was developed about ten years ago. This source term was based on data and calculational tools developed in the early 1980s. Since that time, there have been experimental programs (Marviken, LACE, ACE)[18,19] implemented to better understand the transport and deposition of fission products in ISLOCA conditions. The results of these experiments indicate that the source term used in the Oconee assessment is somewhat conservative. The LACE experiments indicated decontamination factors (DFs) in the range of 5 (for nonhydroscopic material) to 50 (for hydroscopic material). These experimental DFs are higher than that used in the original Oconee ISLOCA assessment. As a result, the source term used in this generic ISLOCA assessment will likely provide a high estimate to the risk of this accident sequence.

The risk associated with the ISLOCA sequences are estimated after the event trees are quantified and they are combined with the appropriate consequence estimates. The consequences estimates can be calculated using a code such as MELCOR Accident Consequence Code System (MACCS).[20]

## 2.8 Sensitivity Studies

A number of ISLOCA issues are examined through sensitivity studies. These sensitivity studies are used to assess an issue's relative influence on CDF and risk. ISLOCA issues can be related to the methods used to perform the evaluations as well as uncertainties in the estimated parameters. For the initial plant evaluations, issues were chosen because (a) there was a relatively large uncertainty in the values used for a particular parameter, (b) a potential fix was postulated that was expected to result in a significant reduction in CDF and risk, or (c) a different means of establishing probabilities was being considered, which could be used for evaluation of future plants. The complete PRA model was used for this re-evaluation process. This not only provided an accurate estimate of the importance of risk (i.e., its influence on the CDF) but also gave an estimate of the importance of the models and modeling assumptions.

# 3. DESCRIPTION OF THE B&W REFERENCE PLANT INTERFACING SYSTEMS

The B&W reference plant modeled in this generic assessment began commercial operations in the late 1970s. The interfacing systems of this unit were used as part of this study. This referenced B&W reactor is designed for a core power level of 2,772 MW(t) and a net electrical output of 906 MW(e).

## 3.1 Interfacing Systems

All interfacing systems were reviewed to identify those systems that required further evaluation. Screening criteria dictated that an interfacing pipe size larger than 1 in. would be evaluated. The discharge from a high-pressure, 1-in. pipe break, is about 200 gpm. Leak rates outside of the containment that exceed 200 gpm may be risk significant because (a) the capacity of the borated water storage tank (BWST) at the reference B&W plant is approximately 480,000 gal, (b) the capacity of a single RCS makeup pump is 150 gpm, (c) the normal makeup rate to the BWST is 150 gpm, and (d) conservative estimates indicate that it would take approximately 10 hours for the plant to achieve cold shutdown.

The screening review also suggested that the high-pressure injection (HPI) discharge lines, the low-pressure injection (LPI) discharge lines, and the decay heat removal (DHR) letdown lines required further evaluation. Figure 3 is a schematic diagram showing the hardware configuration of the HPI system and Figure 4 provides similar information for the DHR/LPI sys[illegible] Additional details on these systems are p[illegible] Appendix C. The HPI interface com[illegible]s four separate reactor pressure vessel (RPV) injection lines. Starting from the RPV, each injection line contains two check valves that are welded together (hence they cannot be individually leak tested), a normally closed MOV, and the HPI pump discharge check valve. The four lines are identified by the associated MOV, namely HP-2A (B, C, D). The HPI A-line is also used for normal RCS makeup by the makeup and purification system (MU&P). The MU&P system connects to the HPI A-line between the two check valves and the normally closed MOV (HP-2A).

## 3.2 Possible ISLOCA Sequences

By examining system interfaces and plant operational information, PRA and human factors specialists developed possible ISLOCA interface sequences. In some cases (i.e., the LPI lines), the sequences are strictly hardware driven; that is, the ISLOCA potential is a function of the hardware failure rates of the PIB valves. In other cases (i.e., the DHR letdown lines), the possible ISLOCA sequences are initiated by human errors. Table 1 summarizes the ISLOCA sequences identified for the B&W plant analysis.

**3.2.1 LPI Sequences.** Only a single ISLOCA sequence was identified for the LPI interface. Because of the ab[illegible]ce of routine operations associated with [illegible] system, this sequence comprises th[illegible] hardware related check valve failu[illegible] [illegible]aracterize the classical V-sequence

**3.2.2 DHR Sequences.** The DHR system is used for removing core decay heat when the plant operates in shutdown modes 4 or 5. This system is connected to one of the RCS hot legs with a 12-in. pipe, which is isolated by two 12-in. motor-operated gate valves in series (DH-12 and DH-11). There is also an 8-in. line that bypasses DH-11 and DH-12. It has two local, manually-operated gate valves in series, which are not instrumented in any way.

There are two possible ISLOCA sequences: (a) the premature opening of the DHR letdown line while the plant is in the process of shutdown but not yet in the operating range of the DHR system (i.e., the RCS pressures and temperatures are above approximately 300 psi and 300°F), and (b) leaving the DHR letdown line open after the DHR operating pressure limit has been exceeded

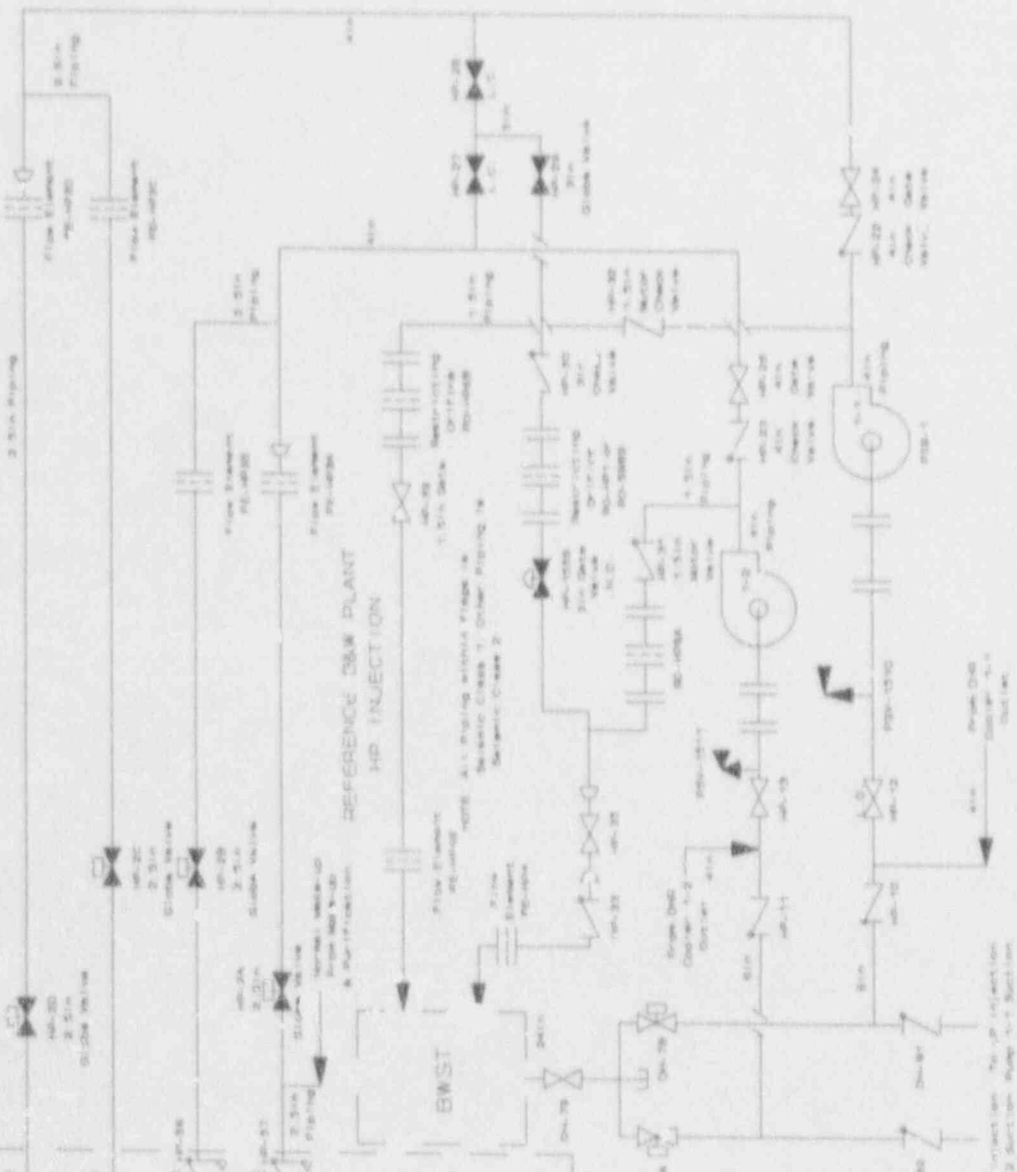


Figure 3. Schematic diagram of the reference B&W plant: HPI system.

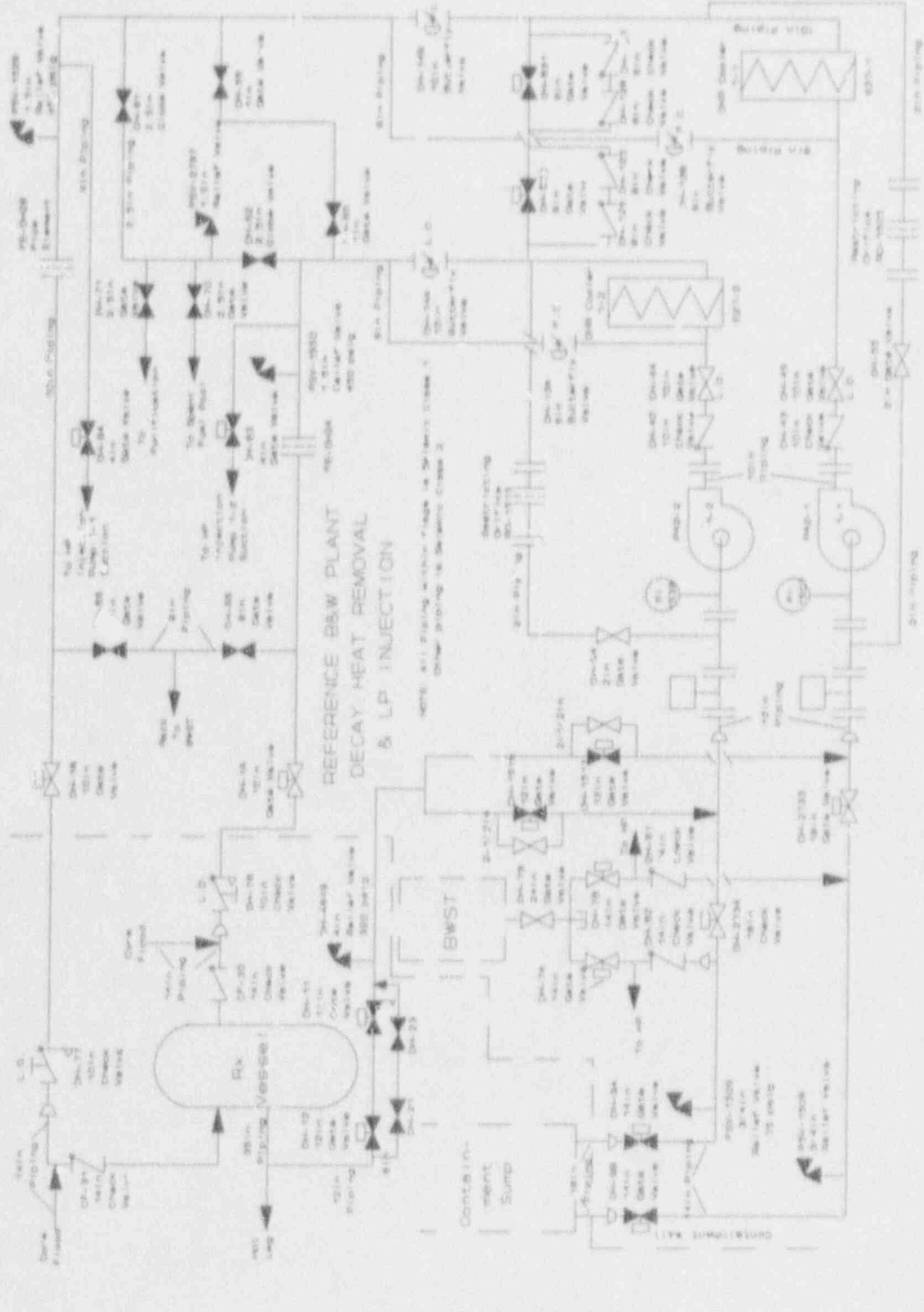

**Figure 4.** Schematic diagram of the reference B&W plant DHR/LPI system.

**Table 1.** List of ISLOCA interface sequences.

| Interface system | Sequence description | Sequence ID |
|---|---|---|
| LPI (two lines) | Hardware failure of two check valves | LPI |
| DHR-letdown (shutdown) | Premature opening of letdown MOVs during shutdown | DHR-SD |
| DHR-letdown (startup) | Startup with letdown MOVs left open | DHR-SU |
| HPI (B, C, and D legs) | Hardware failure of two check valves and stroke test of MOV | HPI |
| HPI (A-leg) | Stroke test of HP-2A and failure of two check valves | MU&P |

during plant startup. In both situations, the DHR system is exposed to high-pressure reactor coolant that could result in the rupture of some low-pressure rated components.

**3.2.3 HPI Sequences.** During most operating modes, the MU&P system provides cleanup of the RCS water and seal injection to the reactor coolant pumps. The normal makeup flows from the MU&P system through the HPI A-header via check valves HP-57 and HP-59.

Several features of the reference plant HPI interface create the potential for an ISLOCA related scenario. These features are as follows:

- The HPI pressure isolation check valves in each injection line (HP-57/59, HP-56/58, HP-48/50, and HP-49/51) are welded together. This arrangement prevents leak testing each valve individually. Therefore, a successful leak test does not necessarily confirm that both of the check valves are properly seated; it is possible that one of the valves might be improperly seated because of improper installation or maintenance.

- The normally closed HPI-MOVs (HP-2A, B, C, and D) are stroke tested quarterly. When the HPI A-header valve (HP-2A) is stroke tested, the MU&P system continues to provide RCS makeup through that line. When HP-2A is opened, high-pressure makeup water backflows to the HPI-pump discharge check valve (HP-23). Once the test is completed, the MOV is closed, and the HPI line is vented to the pump recirculation line between valves HP-29 and HP-1556. The latter valve remains closed during this procedure. However, this same recirculation line is opened to the BWST for the quarterly HPI-pump flow test. This process presents an opportunity for misaligning the recirculation line after the pump test, and/or HP-2A after the stroke test, possibly allowing RCS water to backflow to the BWST.

# 4. B&W REFERENCE PLANT RESULTS

A detailed understanding of the capabilities of the plant hardware and personnel is needed to accurately estimate the probability and consequences of an ISLOCA because of the specific nature of the various sequences. Although the B&W BWST inventory is normally maintained at approximately 480,000 gal, a small ISLOCA (equivalent to a 2-in. line) would result in an initial leak rate of about 1,000 gpm. This leak rate would deplete the BWST in about 8 hours. This depletion time is less than the 10 hours it takes to achieve cold shutdown. Other postulated ruptures, particularly those associated with the DHR system, can result in much larger leakage rates. However, if the rupture is isolated in a timely manner and the leak is terminated, the plant can be safely cooled down using the auxiliary feedwater (AFW) system and steam generators (SG).

Before discussing the individual event trees developed for the B&W plant, some general comments applicable to all the postulated ISLOCA sequences are provided. During the course of the plant visit, particular attention was paid to the issue of local environmental effects resulting from ruptures in the interfacing systems. Because of the probabilistic nature of the calculation, definitive rupture locations can not be pinpointed. Therefore a general survey was made of the interfacing system flow paths to *qualitatively* estimate the impact of ruptures on equipment in various locations. This survey included walkdowns of the emergency core cooling systems (ECCS) to examine the most likely break locations.

This analysis made the assumption that all equipment in the compartment where a postulated ISLOCA rupture occurs will be rendered unavailable for use in isolating/mitigating the ISLOCA. Equipment in compartments judged to be candidate locations for an ISLOCA break was inventoried as a result. The plant survey did not verify the assumption that the ECCS are adequately separated. This verification was performed analytically. Adequate separation is required to ensure that any postulated ISLOCA rupture would not affect redundant trains. The ECCS systems at the reference B&W plant are arranged with all A-train ECCS equipment in one room and all the B-train equipment in a second room. However, this specific design does not preclude the possible propagation of a severe ISLOCA environment to the redundant equipment. It does, however, make it less likely to occur than if the trains were not separated.

One result of the plant survey indicates that there are two design aspects of the DHR system that could influence an induced failure of the DHR redundant train. These two aspects are (a) this specific B&W design includes a single DHR letdown line from the RCS, which becomes low-pressure rated immediately after the second pressure isolation valve (i.e., DH-11), and (b) both DHR heat exchangers are located side-by-side in the same room (Room 113).

This low-pressure line downstream of DH-11 was identified as one of the likely rupture locations. Failure of this letdown line would therefore disable all DHR cooling. This low-pressure line can fail either inside or outside the containment. If the break were to occur inside the containment and if the break was not isolated, low-pressure recirculation from the containment sump could be accomplished. If the break were isolated, recirculation would be unnecessary unless heat removal by the steam generators could not be maintained or re-established. In the event that cooling was not available via the steam generators, it would be necessary to use feed and bleed cooling through the power-operated relief valve (PORV). Because of the different and diverse paths of possible coolant injection, failure of the low-pressure line inside the containment does not significantly influence the risk of the ISLOCA sequence. For the purpose of understanding the risk profile of the ISLOCA sequences, the failure of this line is assumed to be outside of the containment.

The second area of interest with respect to equipment location and redundancy lies with the DHR heat exchangers. The DHR heat exchangers were identified as likely rupture locations for the

DHR and LPI ISLOCA sequences. For this referenced B&W plant, both DHR heat exchangers are located in the same room. Although the heat exchangers themselves would probably not degrade in a harsh environment, there are some air-operated bypass and isolation valves that may not survive the harsh environment that is likely during an ISLOCA. Therefore, if one of the heat exchangers fails, subsequent failure of the other is likely. However, primary system heat removal is still possible under these ISLOCA conditions through the steam generators using the power conversion system.

The plant survey described above was not able to determine if environmental conditions resulting from an ISLOCA sequence would cause failure of redundant ECCS trains. This issue of common cause failure induced by ISLOCA environmental conditions was addressed separately through RELAP5[21] and CONTAIN[22] calculations. For the reference B&W plant of this study, it was determined that high temperature and humidity resulting from an ISLOCA would not be likely to cause equipment failure in the separate and redundant ECCS train. This was the result of the lack of superheating of the auxiliary building's atmosphere and the qualification of the ECCS equipment for high-energy line break conditions. Flooding, however, was identified as a potential common cause failure mechanism.

The NRC is sponsoring a valve testing program at the Idaho National Engineering Laboratory (INEL) as part of the resolution of Generic Issue 87, "Failure of HPCI Steam Line Without Isolation."[23] Preliminary results from this program indicate that the standard calculations performed by industry for sizing motor-driven valve operators might underpredict the force required to open and close MOVs during severe break flows. Each of the reference plant's pressure isolation valves was evaluated using the results of the preliminary valve testing work. The results of these analyses are documented in Appendix C, Section C.5. The program researchers concluded that the reference plant's HPI and LPI isolation valves [HP-2A(B, C, D), and DH-1A(B)], will operate at any RCS pressure likely to occur during an ISLOCA. However, the DHR letdown isolation valves (DH-11 and 12) are not likely to operate above approximately 1,900 psid (the operability threshold). The valve testing program is still ongoing, and some possibly conservative assumptions were made concerning disc friction factors and the differential pressure across the valve. Therefore, the 1,900 psid operability threshold was not factored into the ISLOCA analysis.

[Note: Preventing the valve from being prematurely opened above 1,900 psid would only affect the initiation of the DHR-shutdown sequence thereby reducing the system large-rupture probability from 0.113 to 0.111. Also, once the pressure boundary is breached, the primary system begins to blowdown, resulting (after a period of time) in a reduction of the effective delta-P across the valve. Consequently, for the valves examined in these analyses, it is assumed that if a valve is able to open, it too will be strong enough to close. This assumption in some cases may not be consistent with the original valve design specification. The valve's design information indicates that the valves may be able to open at high pressure, but they may not be able to reclose at pressures above 500 to 600 psid. This design information differs from the new experimental data and calculations described in Appendix C. The significance of this potential inability to reclose the valve depends on the actual pressure at the time of the ISLOCA and on the differential pressure during the blowdown following the rupture. Because of the relatively low probability assessed for entry into DHR at very high pressure and the likelihood that the pressure would be reduced subsequent to the break in the DHR system, the effects of this inability to reclose the valve at high pressure are minimal.]

## 4.1 Component and System Rupture Probabilities

An important part of quantifying the ISLOCA sequences is the estimation of the rupture probabilities and likely failure locations for the interfacing systems exposed to an overpressure condition. Once the most likely accident sequences were identified, those portions of systems being overpressurized were identified and analyzed to calculate the probability of a rupture. The basic process involved the following: (a) estimating the pressure capacities for the components in the interfacing system, (b) estimating the local system pressure generated in the interfacing system as a result of an ISLOCA sequence, and (c) combining these two estimates in a stress/strength comparison to calculate a rupture probability for both the individual components and the entire interfacing system.

Each ISLOCA sequence was examined and the interfacing system reviewed to identify the equipment and subsystems exposed to pressures higher than their rated pressures. The estimation of realistic rupture pressures for the fluid system components that compose the interfacing systems was performed in a separate analysis by Wesley et al.[6]

The Wesley et al. results, the local system pressure estimates, and the rupture probability calculations are described below.

**4.1.1 Pressure Fragility Calculations.** This section briefly summarizes the work performed by Wesley et al. in support of the ISLOCA analysis. The purpose of this work is threefold: (a) develop a methodology to assess the pressure capacity of fluid systems when subject to high-pressure and high-temperature reactor coolant, (b) determine the leak or rupture pressure capacity and associated uncertainty for interfacing system components, and (c) identify the likely leak rates and leak areas when a rupture is predicted.

The methodology is described in *Pressure Dependent Fragilities for Piping Components*[6] and will not be discussed here, except to identify some of the basic assumptions made. Foremost is the intention to generate realistic pressure capacities. This requirement prompts the use of actual material properties and test data rather than code or design specifications. The pressure capacity of an individual component is assumed to be a lognormal random variable. A lognormal distribution is used because it has proven to be a valid description of material properties. The pressure capacity can also be expressed as the product and quotient of several random variables. The central limit theorem states that this type of aggregate tends to be lognormal regardless of the individual distributions. Lastly, the pressure capacities are calculated assuming quasi-static pressure and temperature conditions. This is based on runs of RELAP5 models of the interfacing systems. These RELAP5 models calculate the local system pressure histories after the pressure isolation boundary has been violated.

All major components in the interfacing systems were evaluated. This included pipes (all of which are stainless steel), tanks, vessels, heat exchangers, flanges, valves (both packing and bolted bonnets), and pumps (both the seals and the casing).

The ISLOCA sequences identified for the B&W plant comprise both the HPI and the DHR/LPI systems. The components in these systems were evaluated and the median failure pressures calculated, along with the uncertainty in the failure pressure estimate. The results from this analysis are presented in Tables 2 and 3.

Given a pressure capacity estimate and a measure of the uncertainty in that estimate, it is possible to calculate the failure (i.e., rupture) probability for any internal pressure. The calculation is analogous to that performed for seismic induced failure.[24] Specifically, the probability that the actual failure pressure is below the internal pressure is calculated by standardizing the failure pressure random variable and calculating the value of the standard normal (Gaussian) function. For example, for a 12-in. schedule 20 pipe, the median failure pressure is 1,660 psig, and the uncertainty (expressed as a logarithmic standard deviation or "beta") is 0.36. The failure probability of that pipe given an internal pressure of 2,100 psig, is expressed as

$$\mathrm{Prob}(P_f<P_i) = \Phi((\ell N(P_i) - \ell N(P_f'))/\beta)$$

$$\mathrm{Prob}(P_f<2{,}100) = \Phi((\ell N(2{,}100) - \ell N(1{,}660))/0.36)$$

$$\mathrm{Prob}(P_f<2{,}100) = \Phi(0.65) = 0.742 \qquad (1)$$

**Table 2.** Pressure fragilities of HPI components.

| Component | Description | Median failure pressure (psi) | Logarithmic mean | Logarithmic standard deviation |
|---|---|---|---|---|
| HCC-91 | 3-in. pipe, schedule 10S | 2,712 | 7.905 | 0.36 |
| FE-HP4 | 3-in. 150 psi flow element | 955 | 6.862 | 0.04 |
| HP-33 | 3-in. swing check valve | 5,507 | 8.614 | — |
| P58-2 | HPI pump 1-2 | 2,250 | 7.719 | 0.25 |
| GCB-4 | 6-in. pipe, schedule 10S | 1,644 | 7.405 | 0.36 |
| 6GCB4a | 6-in. 300 psi flange-a | 2,362 | 7.767 | 0.12 |
| 6GCB4b | 6-in. 300 psi flange-b | 2,362 | 7.767 | 0.12 |
| HP-13 | 6-in. 300 psi local-manual valve | 2,170 | 7.682 | 0.25 |
| GCB-2 | 4-in. pipe, schedule 10S | 2,075 | 7.638 | 0.36 |
| GCB-11 | 4-in. pipe, schedule 10S | 2,075 | 7.638 | 0.36 |

**Table 3.** Pressure fragilities of DHR/LPI components.

| Component | Description | Median failure pressure (psi) | Logarithmic mean | Logarithmic standard deviation |
|---|---|---|---|---|
| GCB-7 | 12-in. pipe, schedule 20 | 1,660 | 7.415 | 0.36 |
| DH-1517 | 12-in. motor operated gate valve, 300 psi | 1,704 | 7.441 | 0.20 |
| GCB-8 | 18-in. pipe, schedule 20 | 1,488 | 7.305 | 0.36 |
| DH-2733 | 18-in. motor operated gate valve, 300 psi | 2,277 | 7.731 | 0.20 |
| HCB-1 | 18-in. pipe, schedule 10S | 843 | 6.737 | 0.36 |
| HCB-1 | 14-in. pipe, schedule 10S | 1,090 | 6.994 | 0.36 |
| DH-81 | 14-in. swing disk check valve, 150 psi | 1,445 | 7.276 | 0.20 |
| 12-GCB-8 | Pipe, schedule 20 | 1,660 | 7.415 | 0.36 |

**Table 3.** (continued).

| Component | Description | Median failure pressure (psi) | Logarithmic mean | Logarithmic standard deviation |
|---|---|---|---|---|
| 12GCBa | Flange, 300 psi | 2,250 | 7.719 | 0.12 |
| 12GCBb | Flange, 300 psi | 2,250 | 7.719 | 0.12 |
| 12GCBc | Flange, 300 psi | 2,250 | 7.719 | 0.12 |
| P42-1 | DHR pump 1-1 | 2,250 | 7.719 | 0.20 |
| GCB-1 | 10-in. pipe, schedule 20 | 1,984 | 7.593 | 0.36 |
| 10GCB1a | 10-in. flange, 300 psi | 2,485 | 7.818 | 0.12 |
| DH-43 | 10-in. swing disk check valve, 300 psi | 2,016 | 7.609 | 0.20 |
| DH-45 | 10-in. hand wheel-operated globe valve, 300 psi | 2,170 | 7.682 | 0.2[illegible] |
| E271T | DHR heat exchanger tube sheet | 432 | 6.068 | [illegible] |
| E271P | DHR heat exchanger plastic collapse | 1,030 | 6.937 | 0.23 |
| E271C | DHR heat exchanger cylinder rupture | 1,630 | 7.396 | 0.27 |
| E271A | DHR heat exchanger asymmetric head buckling | 2,030 | 7.616 | 0.23 |
| E271a | 10-in. out-f, 300 psi | 2,485 | 7.818 | 0.12 |
| E271b | 10-in. in-f, 300 psi | 2,485 | 7.818 | 0.12 |
| GCB-10 | 6-in. pipe, schedule 10S | 1,585 | 7.368 | 0.36 |
| GCB-10 | 10-in. pipe, schedule 20 | 1,984 | 7.593 | 0.36 |
| GCB-10 | 8-in. pipe, schedule 20 | 2,503 | 7.825 | 0.36 |
| DH-128 | 8-in. swing disk check valve, 300 psi | 1,242 | 7.124 | 0.20 |
| GCB-2 | 4-in. pipe, schedule 10S | 2,075 | 7.638 | 0.36 |
| FE-DH2B | 10-in. flow element, 300 psi | 2,485 | 7.818 | 0.12 |

where

$P_f$ = actual failure pressure

$P_i$ = local-internal pressure

$P_f'$ = median estimated failure pressure

$\beta$ = logarithmic standard deviation of $P_f$ [standard deviation of the corresponding normal distribution of $P_f$, i.e., standard deviation of $\ell N(P_f)$]

$\Phi(\ )$ = standard normal (Gaussian) function

$\ell N(\ )$ = natural logarithm.

**4.1.2 Estimating Local System Pressures.** Because the interfacing systems contain either relief valves or are open-ended (i.e., connected to the BWST, which is vented to the atmosphere), exposing them to the high-pressure RCS results in flow through the system (even before any ruptures occur). Consequently, the pressure drops (sometimes significantly) for components further away from the RCS. Estimating the internal pressures produced in the interfacing systems was performed by building and running RELAP5 models of the interfacing systems. Each of the five ISLOCA sequences postulated (two HPI and three DHR/LPI) was represented with a RELAP5 model. Only the interfacing system was modeled in detail. The boundary parameters (i.e., RCS conditions) were modeled as a constant pressure and temperature. This simplification is only slightly conservative because of the rapid pressurization of the interfacing system once the pressure isolation valves begin to open. The pressure in the interfacing system reaches equilibrium within approximately 5 to 7 seconds. The pressure reduction of the primary system resulting from the blowdown of the primary system during this time is likely to be on the order of 200 to 300 psi. This pressure reduction will have only a small impact on the potential for failure of an interfacing systems component. Figure 5 shows the steady-state pressure drop through the DHR, as a function of RCS pressure. The effect of the relief valves in the DHR system produces a maximum pressure drop of from 2,200 psi to about 1,400 psi for certain portions of the system. Figure 6 shows the time-dependent pressure at various points in the DHR system for an RCS pressure of 2,200 psi. Both of these graphs do not include the effect of any ruptures in the system, only the existing relief valves.

**4.1.3 HPI-Sequences Rupture Probability.** Although they are different in their initiation and sequence frequencies, the two HPI sequences (identified as HPI and MU&P elsewhere in this report) are basically identical with respect to the local system pressures generated and the expected response of the system components.

The HPI ISLOCA sequence is initiated by the stroke test of the HPI injection valve (HP-2B/C/D, see Figure 3) in conjunction with the existing leakage of the pressure isolation check valves (HP-56/58, HP-48/50, and HP-49/51). However, in order to threaten lower pressure-rated equipment, additional failures are required. Specifically, either the HPI pump discharge check valve (HP-23/22) must fail to close or the vent line to the BWST must be inadvertently left open (HP-27/26 and HP-29). These two additional failures lead to the identification of o specific rupture scenarios, which are characterized by an overpressure in the HPI pump suction line or in the BWST vent line. These two scenarios were evaluated by building and running a RELAP5 model to estimate the local system pressures generated by an ISLOCA sequence.

The BWST vent-line scenario results in RCS water backflowing through the HPI injection line, to the HPI full-flow recirculation line, and into the BWST. The recirculation line contains an orifice (RO-HP1) that is designed to restrict flow to the equivalent of one HPI pump (i.e., about 500 gpm). Modeling this flowpath using RELAP5, and assuming that the scenario was initiated by opening the injection MOV (opening time 10 seconds and check valves stuck in full open position), produced a local system pressure

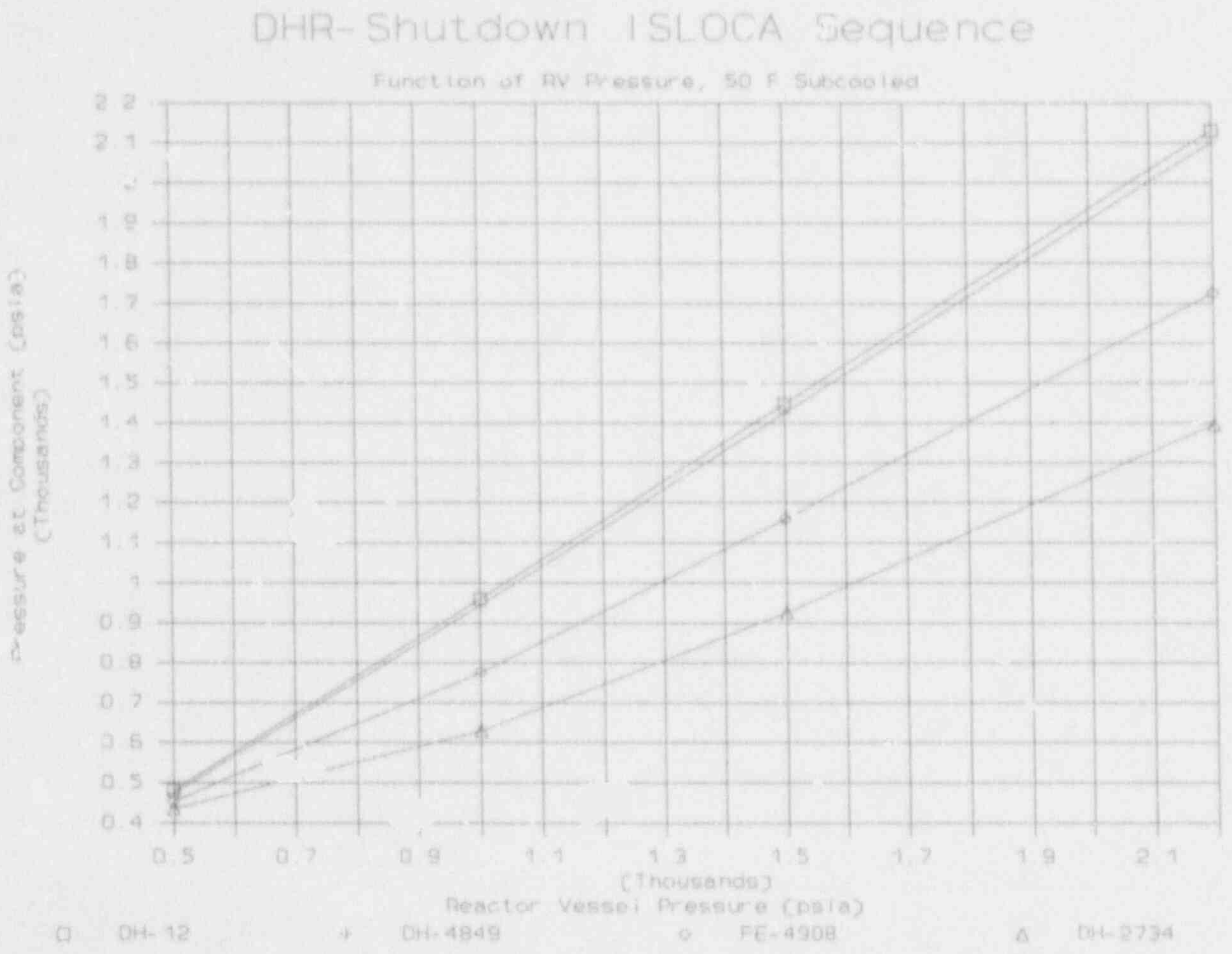


**Figure 5.** Pressure drop through DHR system as a function of RCS pressure.

estimate of 650 psig on the BWST side of the restricting orifice RO-HP1.

**4.1.4 DHR/LPI Sequences Rupture Probability.** Three sequences compose the DHR/LPI category: (a) DHR-shutdown (SD), (b) DHR-startup (SU), and (c) LPI sequences. While each involves some unique features concerning their initiation and frequencies, they all result in an overpressure in the DHR/LPI system (the same system is used for both DHR and LPI operations). Therefore, even though some differences exist in the postulated RCS conditions for each sequence, the assumption is made that these sequences result in the overpressure of the same components. One small exception to this lies in the LPI sequence, where the core flood tank (CFT) scenario results in the overpressure of the CFT (see LPI event tree for more information).

The LPI sequence is postulated to occur at full RCS operating pressure (about 2,265 psi) and temperature (about 600°F). At these conditions, the LPI system is assumed to rupture with a probability of one. (The CFT scenario is considered separately, see description in Appendix D). The DHR-SU sequence is postulated to occur such that the operators have an opportunity to recover the sequence before rupturing the DHR system. Therefore, by definition, the DHR-SU sequence "initiation" is assumed to progress to the point where a rupture occurs, which results in an effective rupture probability of one. Only in the DHR-SU sequence is there uncertainty about the rupture probability of the DHR system when subject to an overpressure situation. This results from the uncertainty in the RCS pressure at the time of the premature opening of the DHR letdown isolation valves. Specifically, it is difficult to support a

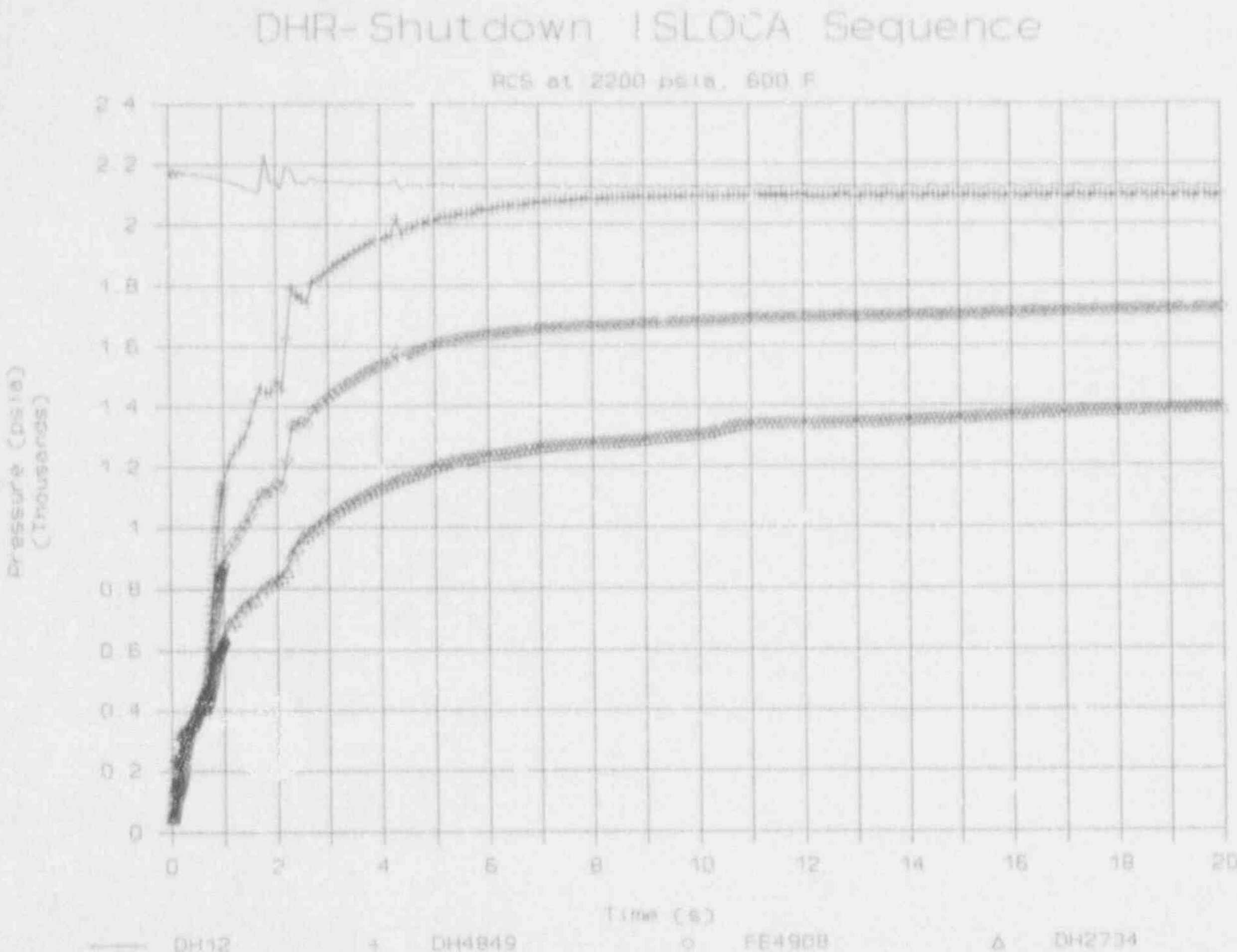


**Figure 6.** DHR system pressure response to DHR-SD ISLOCA.

prediction of the RCS pressure at which the operators might inappropriately (i.e., prematurely) enter into DHR cooling. Consequently, a method was used to weight the pressure-dependent system rupture probabilities based on the HEP (which was also assumed to be a function of the RCS pressure) of prematurely opening the DHR letdown MOVs (DH-11 and DH-12). This was done by developing a discrete probability distribution for the HEP for premature opening of the DHR isolation valves. This RCS-pressure dependent probability density is shown in Figure 7 [this is *not* the HEP but a probability density function of the HEP (i.e., a probability of a frequency)]. Figure 8 displays the aggregated system rupture probabilities (for large ruptures, small leaks, and no failures) for the DHR/LPI system as a function of RCS pressure (not local system pressure). Table 4 lists the individual component failure pressures (median values) along with an average failure probability (i.e., averaged over RCS pressures from 300 to 2,200 psi).

## 4.2 Human Reliability Analysis

**4.2.1 Introduction and Overview.** This section summarizes the results of the ISLOCA HRA efforts. Appendix E provides detailed information regarding HRA fault trees, event trees, tabulated HEP values, and discussions of the HRA process. HEPs presented as part of the HRA analysis are estimates based upon contemporary models and quantitative techniques. As in any HRA, these HEPs are not intended to stand alone since they are multiplied by hardware failure uncertainties in calculations for CDFs. Therefore, individual HEPs should not be used in isolation since they

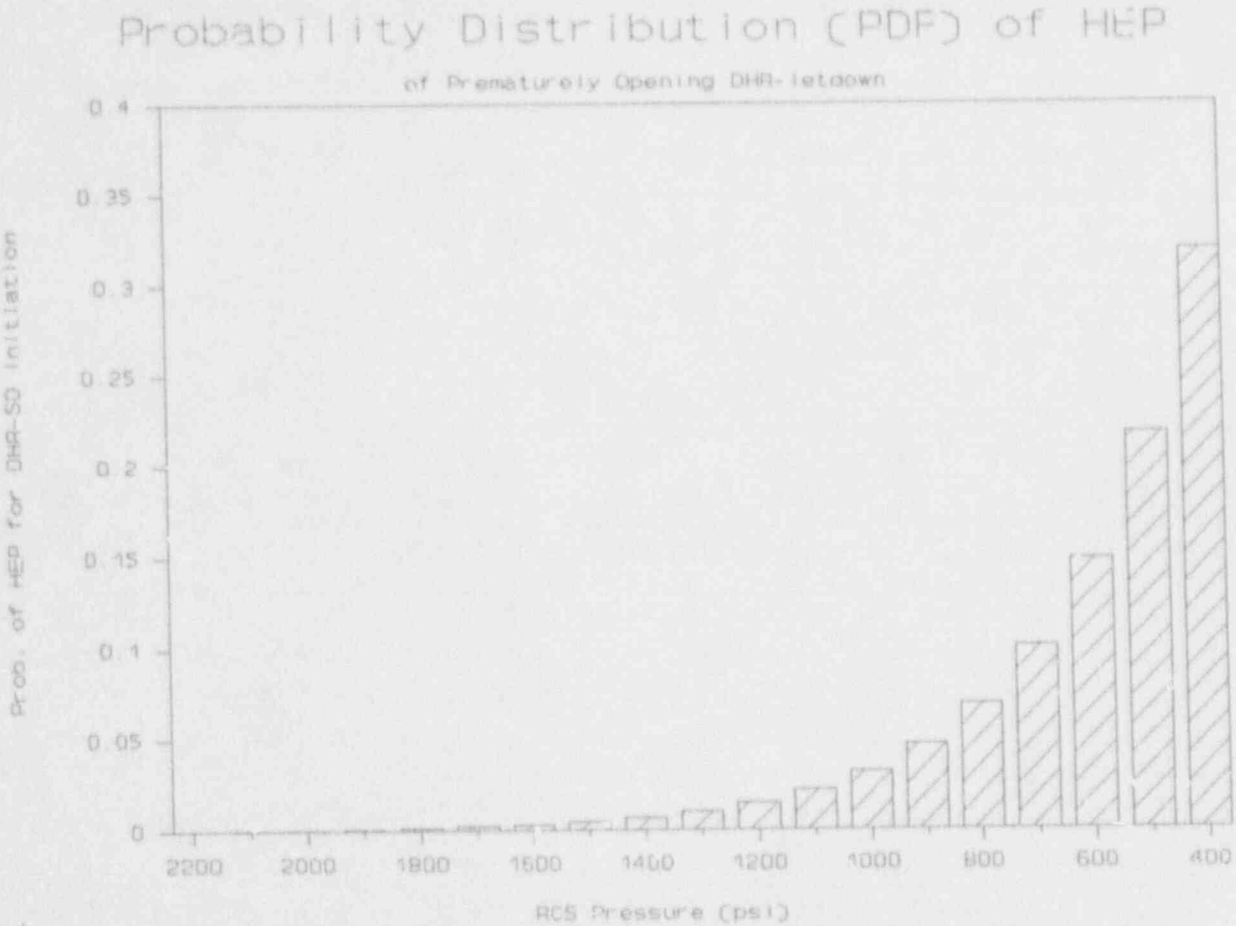


**Figure 7.** Probability density function of the HEP of prematurely entering DHR cooling.

must be considered in the context of a specific scenario, as well as the hardware failure information contained within this report.

HRA was used to model the important human errors for each scenario in the ISLOCA PRA. As discussed in Section 2.5, HRA is a methodological tool that involves the quantitative analysis, prediction, and evaluation of work-oriented human performance. The ISLOCA HRA diagnosed those factors within the plant's systems that could lead to less-than-optimal human performance in the initiation, detection, diagnosis, and mitigation of ISLOCA scenarios. HRA was used as a diagnostic tool to isolate the error rate anticipated for individual tasks and to determine where errors were likely to be most frequent.

Because most of the human actions in this HRA involved the use of various normal, abnormal, and emergency operating procedures, THERP-type HRA event trees were chosen for modeling most of the human actions in the detailed analysis. However, in several ISLOCA scenarios, HRA fault trees were used in conjunction with the typical THERP event trees to provide the best representation of the modeled events. Detailed analyses were conducted using the fault trees and/or THERP event trees to estimate the error probabilities of the dominant human actions.

Individual error branches for each of the HRA event trees (see Section 2.5 or Appendix E for details) were quantified using techniques from THERP, NUCLARR,[11] and engineering judgement. Specific human actions on each error branch were assigned an estimate of the basic HEP. These basic HEP estimates were then modified using PSFs to realistically describe the work

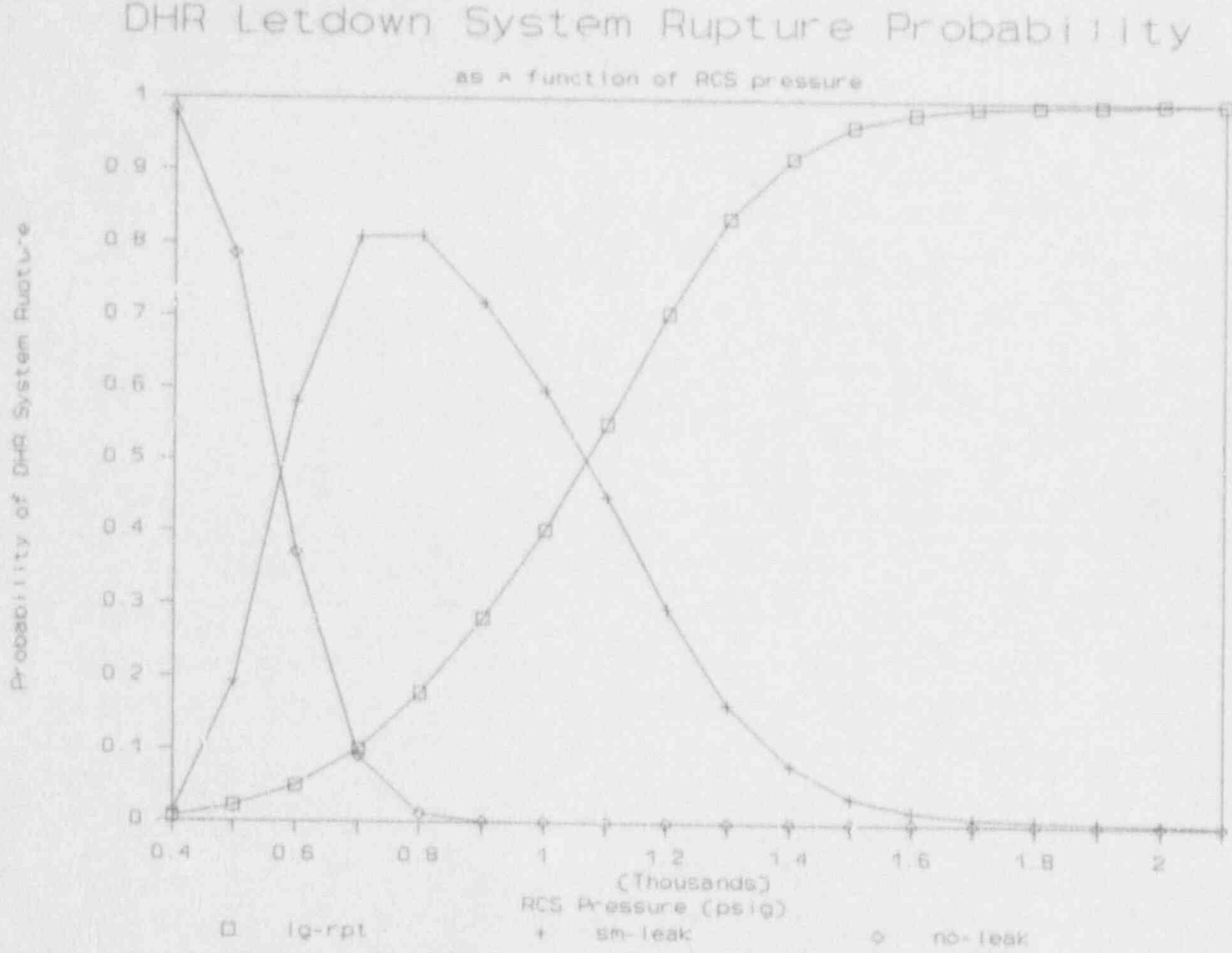


**Figure 8.** DHR/LPI system rupture probability as a function of RCS pressure.

process at the plant. Finally, possible failure paths (i.e., sequences that included either single or multiple human errors leading to a failure of the action modeled by the HRA tree) were identified and combined to estimate the total failure probability for the HRA tree, in accordance with the THERP guidelines. Individual PSFs were derived from task analyses, time-line analyses, evaluation of the human/machine interface, and direct observations of operator performance. The majority of these PSFs were presented in the *ISLOCA Inspection Report* for the analyzed plant.[12] Each PSF was seen as casting either a positive or negative influence on the basic HEP, that is, as either decreasing or increasing the probability of failure for a given human action. For example, some of the positive PSFs in evaluations of the B&W plant included the following:

- Workload alone was insufficient to introduce either initiating events or precursors for ISLOCA

- Newly introduced operating schematics (color coded P&IDs with alarms and instrumentation highlighted) could prove to be useful operator aids

- Operators' practice of repeating verbal instruction increases the probability for effective oral communication

- The presence of consistent labeling in the control room contributes to positive operator performance.

Negative PSF findings include the following:

- Lack of specific training on ISLOCA

**Table 4.** Predicted failure probabilities for DHR/LPI components.[a]

| Component | Description | Median failure pressure (psi) | F· ilure probability |
|---|---|---|---|
| DH-4849 | Relief va! | n/a | n/a |
| GCB-7 | 12-in. pipe, schedule 20 | 1,660 | 2.553E – 01[b] |
| DH-2734 | ·°-in. motor operated gate valve, 300 psi | 2,277 | 5.0E – 04 sm |
| DH-1517 | 12-in. motor operated gate valve, 300 psi | 1,704 | 1.3E – 02 sm |
| GCB 8 | 18-in. pipe, schedule 20 | 1,488 | 1.072E – 01[b] |
| DH-273.' | 18-in. motor operated gate valve, 300 psi | 2,277 | 5.0E – 04 sm |
| HCB-1 | 18-in. pipe, schedule 10S | 843 | 4.47E – 01[b] |
| HCB-1 | 14-in. pipe, schedule 10S | 1,090 | 2.695E – 01[b] |
| DH-81 | 14-in. swing disk check valve, 150 psi | 1,445 | 6.75E – 02 sm |
| GCB-8 | 12-in. pipe, schedule 20 | 1,660 | 7.12E – 02 |
| 12GCBa | Flange, 300 psi | 2,250 | 0 |
| 12GCBb | Flange, 300 psi | 2,250 | 0 |
| 12GCBc | Flange, 300 psi | 2,250 | 0 |
| P42-1 | DHR pump 1-1 | 2,250 | 3.0E – 04 sm |
| GCB-1 | 10-in. pipe, schedule 20 | 1,984 | 3.15E – 02 |
| 10GCB1a | 10-in. flange, 300 psi | 2,485 | 0 |
| DH-43 | 10-in. swing disk check valve, 300 psi | 2,016 | 2.5E – 03 sm |
| DH-45 | 10-in. hand wheel-operated globe valve, 300 psi | 2,170 | 9.0E – 04 sm |
| E271T | DHR heat exchanger tube sheet | 432 | 8.546E – 01 (50% sm)[b] |
| E271P | DHR heat exchanger plastic collapse | 1,030 | 5.988E – 02 |
| E271C | DHR heat exchanger cylinder rupture | 1,630 | 4.48E – 02 |
| E271A | DHR heat exchanger asymmetric head buckling | 2,030 | 9.2E – 04 sm |
| E271a | 10-in. out-f, 300 psi | 2,485 | 0 |
| E271b | 10-in. in-f, 300 psi | 2,485 | 0 |
| GCB-10 | 6-in. pipe, schedule 10S | 1,585 | 3.22E – 02 |
| GCB-10 | 10-in. pipe, schedule 20 | 1,934 | 2.95E – 02 |
| GCB-10 | 8-in. pipe, schedule 20 | 2,503 | 7.3E – 03 |
| DH-128 | 8-in. swing disk check valve, 300 psi | 1,242 | 1.42E – 01 sm |
| GCB-2 | 4-in. pipe, schedule 10S | 2,075 | 2.2E – 02 |
| FE-DH2B | 10-in. flow element, 300 psi | 2,485 | 0 |

a. Based on a median RCS pressure of 1,250 psi (uniformly distributed between 300 and 2,200 psi). This primary system distribution is presented for informational purposes and was not used in the analysis. Based on this, RELAP5 modeling predicted a median system pressure at DH-4849 of 1,188 psia and a median system pressure at DH-2734 of 818 psia.

b. Indicates a dominant contributor to the system rupture probability.

- Lack of proper ISLOCA notes, cautions, and warnings in procedures
- Lack of awareness that the computer high-pressure alarm on the HPI line could be caused by either leaking check valves or by the MU&P system operation
- Lack of a valve status board in the control room and absence of procedures for acknowledging computerized alarms.
- No main control board alarm or pressure indication was observed for the DHR system.
- Tagging was mixed, good in some areas and not as consistent in others.

For purposes of this HRA, stress level was considered optimal with three exceptions: (a) when personnel were sent into containment, (b) when personnel were attempting to isolate the ISLOCA, or (c) when site evacuation was said to occur. THERP procedures allow for modifying HEP values as a function of stress level and where such modifications are made they are noted.

A detailed HRA was conducted for each of the significant scenarios identified in this ISLOCA PRA. In addition, an uncertainty analysis was done using IRRAS.[14] The uncertainty analysis provided both median and upper bound HEP estimates, which were used to calculate error factors (EFs) for respective HEPs. Tables 5 through 9 summarize the results of these analyses, which are described in Appendix E. These tables provide the identifier and description for each significant human error, as well as the mean HEPs and error factors for each human action. A lower bound on the failure rate for human actions was assumed to be 1.0E − 04. This lower bound estimate included the possibility of recovery actions by other crew members, and models situations where there is relatively a long time to respond (hours) and to recover from the abnormal event.

**Table 5.** HPI scenario involving quarterly stroke test for 2A, MU&P flow.

| Identifier | Human action | Mean HEP (EF) |
|---|---|---|
| HV1-MU | HP vent line open | 0.0013 (2.94) |
| HM1-MU | HP MOV2A opened for test | 1.0 |
| HM2-MU | Operators fail to close HP MOV2A | 0.008 (2.27) |
| HV2-MU | HP vent line open (per procedure) | 1.0 |
| HD2-MU | Operators fail to detect ISLOCA | 0.0028 (7.4) |
| HDA2-MU | Operators fail to diagnose ISLOCA | 0.006 (14.93) |
| HI2-MU | Operators fail to isolate ISLOCA | 0.002 (3) |

**Table 6.** HPI scenario involving quarterly stroke test, no MU&P flow.

| Identifier | Human action | Mean HEP (EF) |
|---|---|---|
| HM1-HP | HP MOV2B opened for test | 1.0 |
| HV1-HP | HP vent line open | 0.0013 (2.94) |
| HD2-HP | Operators fail to detect ISLOCA | 0.0014 (9.5) |
| HDA2-HP | Operators fail to diagnose ISLOCA | 0.006 (14.93) |
| HI2-HP | Operators fail to isolate ISLOCA | 0.002 (3) |

**Table 7.** Shutdown scenario involving premature opening of DH11 and DH12.

| Identifier | Human action | Mean HEP (EF) |
|---|---|---|
| DM1-SD | Operators open DH11 and 12 too soon | 0.00066 (10.01) |
| DD2-SD | Operators fail to detect ISLOCA | 0.0002 (10.79) |
| DDA1-SD | Operators fails to diagnose ISLOCA | 0.006 (14.93) |
| DI2-SD | Operators fail to isolate ISLOCA | 0.008 (5) |

**Table 8.** Startup scenario involving DHR system.

| Identifier | Human action | Mean HEP (EF) |
|---|---|---|
| DM1-SU | DH-11/12 (MOVs) left open | 0.0002 (3.53) |
| DD1-SU-A,C | Operator fails to detect overpressure given relief valve opens | 0.0001 (16.4) |
| DI1-SU-C,D | Operators fails to isolate RCS (MOVs left open) | 0.0092 (3.0) |
| DM2-SU | DH-21/23 (manual valves) left open | 0.0002 (4.85) |
| DI1-SU-A,B | Operators fail to isolate RCS (manual valves left open) | 0.013 (2.37) |
| DD1-SU-B,D | Operator fails to detect overpressure, given relief valve closed | 0.001 (3) |
| DD2-SU | Operator fails to detect ISLOCA | 0.0001 (22.99) |
| DA1-SU-A | Operator fails to diagnose ISLOCA | 0.52 (1.6) |
| DA1-SU-B | Operator fails to diagnose ISLOCA | 0.59 (1.5) |
| DA1-SU-C | Operator fails to diagnose ISLOCA | 0.29 (2.5) |
| DA1-SU-D | Operator fails to diagnose ISLOCA | 0.43 (1.9) |
| DI2-SU-A,B | Operator fails to isolate ISLOCA | 0.113 (4.26) |
| DI2-SU-C,D | Operator fails to isolate ISLOCA | 0.016 (2.99) |

**Table 9.** LPI system ISLOCA scenario.

| Identifier | Human action | Mean HEP (EF) |
|---|---|---|
| LD2-CFT | Operators fail to detect ISLOCA | 0.0001 (2.05) |
| LDA2-CFT | Operators fail to diagnose ISLOCA | 0.0001 (43.37) |
| LI2-CFT | Operators fail to isolate ISLOCA | 0.149 (5) |
| LD2-LP | Operators fail to detect ISLOCA | 0.0035 (11.15) |
| LDA2-LP | Operators fail to diagnose ISLOCA | 0.01 (10) |
| LI2-LP | Operators fail to isolate ISLOCA | 0.148 (5) |

Inspection of the data reveals that failure rate probabilities are highest for mitigation, isolation, and errors of commission such as inadvertent valve lineup after test, or faulty decisions such as early entry into DHR cooldown. Diagnosis errors range on the order of 5.9E − 01 to 6.0E − 03 and, in many cases, reflect the large amount of time available for the crew to reach an opinion on the event. Rates for isolation and mitigation were observed to be 2.0E − 03 and 1.5E − 01, respectively, and reflect the lack of resources available to crews. These resources, if present would have decreased the failure rate estimates, include an ISLOCA procedure, training on ISLOCA, instrumentation, and a procedure for computer alarm acknowledgement. Without these items, crews could be forced to operate in a knowledge-based realm during an ISLOCA.

Table 10 presents latent errors identified during conduct of the HRA. Each of the errors is preceded by the event sequence number and is followed by the nominal (detailed) HEP value. Description of the error and its sequence are presented in Appendix D.

Only one error of commission was identified as an initiator (i.e., DM1-SD): operators open DH-11/12 too soon in the shutdown cycle. Latent errors involving vent line configuration shown in Table 10 can be of either the omission or commission type. The low failure rate for DM2-SU reflects the double verification for these valves as called out by both procedures SP-03130, "Decay Heat Removal System Isolation Test," and OP-00008, "Operation and Control of Locked Valves."

**4.2.2 Detailed Breakdown of Human Error Actions.** Table 11 represents the distribution of errors modeled in support of ISLOCA evaluation at a B&W plant. The tabled values include all errors modeled in the supporting fault trees and HRA event trees.

As these data indicate, the majority of error types appearing in the present analysis fall into the omission category. This is in keeping with contemporary PRA. What is unique about this ISLOCA PRA is that some 20% of the human errors modeled are from commission and complex commission decision-based sources. Although caution should be taken when extrapolating from one plant's data, these results *do* indicate that existing PRAs may significantly under-represent human contribution to systems failure.

**4.2.3 Decision-Based Errors.** The rates for decision-based errors presented in Table 12 were derived using THERP and engineering judgement techniques. *While these failure rates apply to those decision-based errors identified and quantified in the B&W ISLOCA analysis, they are not limited to instances where the action is the top-level action in an event sequence.* To learn more about where a particular decision-based failure fits within an action flow, the sequence identifiers, task descriptions, failure rates, and EFs are presented in Table 12 (see Appendices D and E for more detail).

**4.2.4 Human Factors Influence on the Risk Associated with ISLOCA.** The current analysis indicates that human errors, particularly, errors of commission, are contributors to the CDF for ISLOCA sequences. However, it is premature at the present time to say whether, in Reason's terminology,[25] "active" errors such as the decision to prematurely enter DHR, or the human contribution to risk due from "latent" errors will be important at other plants. In the present case, both of these types of errors of commission played a significant role in assessing the plant's susceptibility to ISLOCA. If training for ISLOCA had been available at the plant and if personnel had thorough ISLOCA procedures, then the probability for ISLOCA would be reduced. Proceduralizing crew response to computer alarms and providing additional indication of valve status would also reduce risk.

## 4.3 Reference B&W Plant Event Trees

The following sections describe the event trees developed for the five ISLOCA sequences. The quantification of the event trees is based on a yearly timeframe. This is reflected in the frequency of the initial event-tree event. This

**Table 10.** Latent errors.

| Sequence | Event description | Mean HEP |
|---|---|---|
| HV1-MU | HP vent line left open | 0.0013 |
| HV1-HP | HP vent line left open | 0.0013 |
| DM1-SU | MOVs DH 11 and 12 left open | 0.0002 |
| DM2-SU | Local valves DH 21 and 23 left open | 0.0002 |

**Table 11.** Distribution of errors from supporting analyses.

| | Omission | Simple commission | Decision-based |
|---|---|---|---|
| Frequency | 100 | 17 | 13 |
| Percent (%) | 77% | 13% | 10% |

**Table 12.** Decision-based errors (either task or subtask values).

| Identifier | Description | HEP (EF) |
|---|---|---|
| HDA2-MU, HP[a] | ROs fail to conclude ISLOCA (from prior tasks) | 0.006 (14.93) |
| HI2-MU, HP | ROs fail to isolate HP2A, undo what was just done | 0.002 (3) |
| DM1-SD | ROs decide on early entry into DHR | 0.00066 (10.01) |
| DDA1-SD[a] | ROs fail to conclude ISLOCA from event signature | 0.006 (14.93) |
| DI2-SD[a] | Crew fails to send instrument and control technicians to remove jumpers (total HEP = 9.0E−05) | 0.008 (5) |
| DM2-SU | ROs fail to close DH21 and 23 | 0.0002 (4.85) |
| DA1-SU-A | ROs fail to recognize ISLOCA from event signature (local valves open; relief valve opens) | 0.52 (1.6)[b] |
| DA1-SU-B | ROs fail to recognize ISLOCA from event signature (local valves open; relief valve fails closed) | 0.59 (1.5)[b] |
| DA1-S' | ROs fail to recognize ISLOCA from event signature (MOVs open; relief valve opens) | 0.29 (2.5)[b] |
| DA1-SU-D | ROs fail to recognize ISLOCA from event signature (MOVs open; relief valve fails closed) | 0.43 (1.9)[b] |
| LDA2-CFT[a] | ROs fail to conclude ISLOCA-core after rupture | 0.0001 (43.37) |
| LDA2-LP | ROs fail to conclude ISLOCA from past rupture information | 0.01 (10) |

a. Indicates subtask values.

b. Indicates engineering judgement used to estimate HEP.

"initiating event" simply postulates a particular operating mode or status of the plant and includes consideration of multiple interface lines. The plant operating status modeled in the initial event is only slightly conservative. The event trees are based on the plant operating all four quarters per year but also include one outage (during which manual values DH-21 and DH-23 are opened to allow MOVATS testing of DH-11 and DH-12) and with a single startup and shutdown. The event trees are constructed to show the downward branch depicted as the failure event listed at the top of the event tree and the upward branch as the complement of the event (i.e., typically success). The top events are a combination of the individual component failures, human errors, and functional failures that were deemed most appropriate for describing the individual ISLOCA scenario progression.

All event tree quantification described in the main part of this report has been performed on a point estimate basis. Uncertainty bounds for the sequence frequencies have been calculated and are reported in Appendix L. Furthermore, sensitivity studies were performed for a number of issues that are believed to dominate the risk or are thought to possess significant uncertainty.

Finally, each event tree end-state was assigned to one of the following consequence bins:

- **OK**—No overpressurization of the low-pressure system occurred.

- **OK-op**—Scenario results in overpressurization of the interfacing system but the system does not rupture or leak.

- **LK-ncd**—Scenario results in a rupture in, and RCS leakage from, the interfacing system, but no core damage occurs because the leak is either isolated before core uncovery or the leak is too small to jeopardize core cooling.

- **LOCA-ic**—A particular scenario that results in a loss-of-coolant inside containment. These events are not examined in detail in this report and are quantified only to their initiating event frequencies.

- **REL-mit**—An ISLOCA with core damage occurs but the radioactive release is mitigated through some aspect of the accident itself (e.g., the release is submerged) or through some accident management strategy (e.g., actuation of fire protection sprinkler systems).

- **REL-lg**—An ISLOCA with core damage occurs and results in a large unmitigated radioactive release.

The REL-mit and REL-lg bins are sometimes subdivided according to failure location, with the new bins identified as RL1, RL2, etc. These bins are described further in the appropriate sequence description.

**4.3.1 Makeup and Purification System Interface Event Tree—MU&P.** A schematic diagram of the interface between the MU&P and the RCS is shown in Figure 9. The base case ISLOCA event tree for this system is shown in Figure 10. During most operating modes, the MU&P system supplies high-pressure purified makeup to the RCS and seal injection to the reactor coolant pumps. The normal RCS makeup flows from the MU&P system through the HPI A-header via check valves HP-57 and HP-59. The MU&P/HPI system has the following features:

- The HPI pressure isolation check valves (PIVs HP-57/59, HP-56/58, HP-48/50, and HP-49/51) are welded together. This prevents leak testing of individual check valves. Therefore, upon completion of a successful leak test, only one of the two check valves can be assured of being properly seated.

- The normally closed HPI MOVs (HP-2A, B, C, and D) are stroke tested quarterly. While the A-header valve (HP-2A) is being stroke tested, the MU&P system continues to provide RCS makeup through that line. When HP-2A is opened during the test, high-pressure makeup water backflows to the HP-pump discharge check valve (HP-23). Once the test is completed, the HP-2A

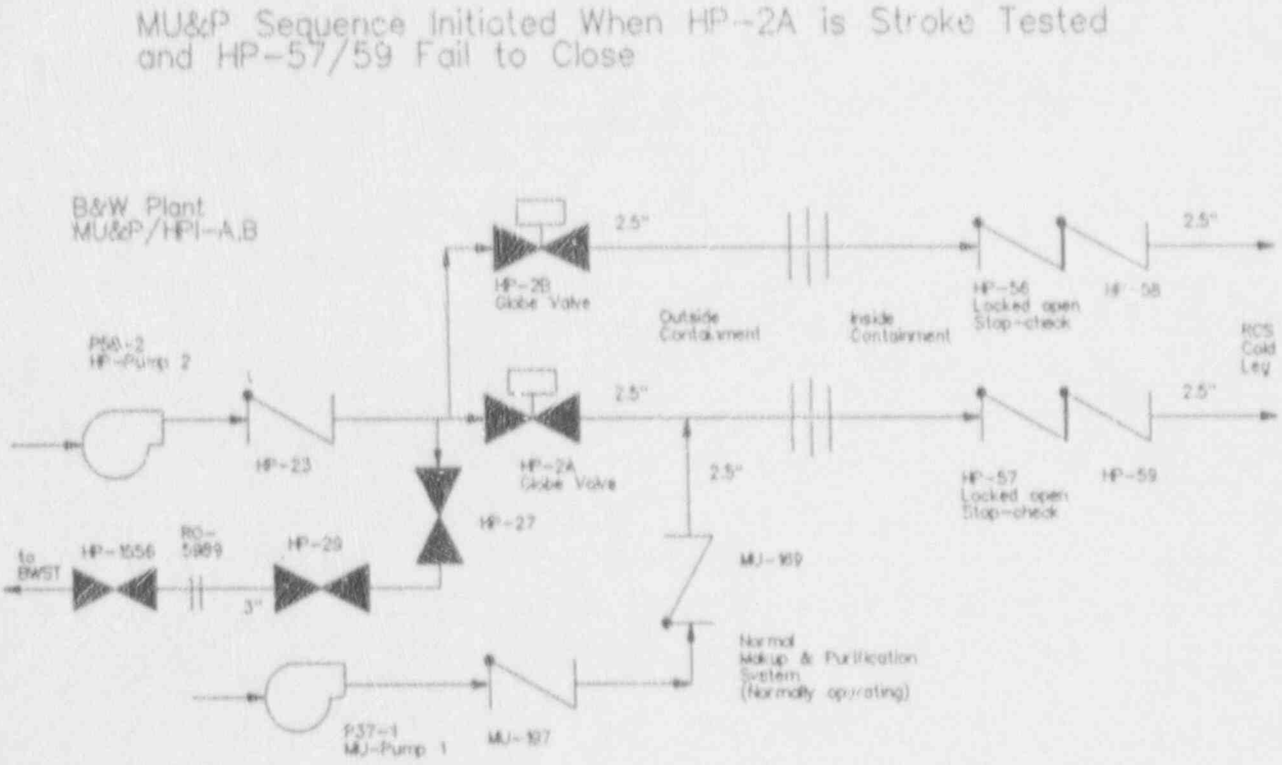


**Figure 9.** Schematic of the MU&P system interface.

MOV is closed, and the HP line is vented to a portion of the HPI-pump recirculation line. This same recirculation line is opened to the BWST for the quarterly HPI-pump flow test. This process presents an opportunity for misaligning the recirculation line after the pump test, and/or HP-2A after the stroke test, possibly allowing RCS water to backflow to the BWST.

The features described above suggest two basic scenarios (i.e., primary coolant flowing to the BWST or to the HPI pump suction pipe). Each one is characterized by unique rupture probabilities and locations. However, the basic timing is assumed to be similar. Bounding calculations were performed to estimate the time to core uncovery for a DHR/LPI and a HPI sequence (see Appendix G). The HPI calculation produced a minimum time to core uncovery, given a rupture in the HPI system, of about 4 hours. Other calculations indicate this minimum core uncovery time might approach 8 hours depending on system performance and operator actions. This difference of 4 hours does not alter the estimate of the HEPs. The 4-hour uncovery time is utilized in the post-rupture recovery event HRA. The individual scenarios determine whether the RCS backflows to the BWST or through the HPI pump. The BWST is vented and therefore cannot be pressurized and the reactor coolant will eventually overflow the tank. The probabilistic rupture calculations (see Listings 6 and 11 in Appendix H) indicate that there is also a 1.0E−04 probability that the pipe leading to the BWST will rupture (note that all pipe ruptures are assumed to be uncontrolled or large ruptures). If the HPI pump discharge check valve fails to close, then the pump suction piping will be pressurized. This scenario results in a system rupture probability of 92% (see Listing 22 of Appendix H), with the 6-in. pipe on the

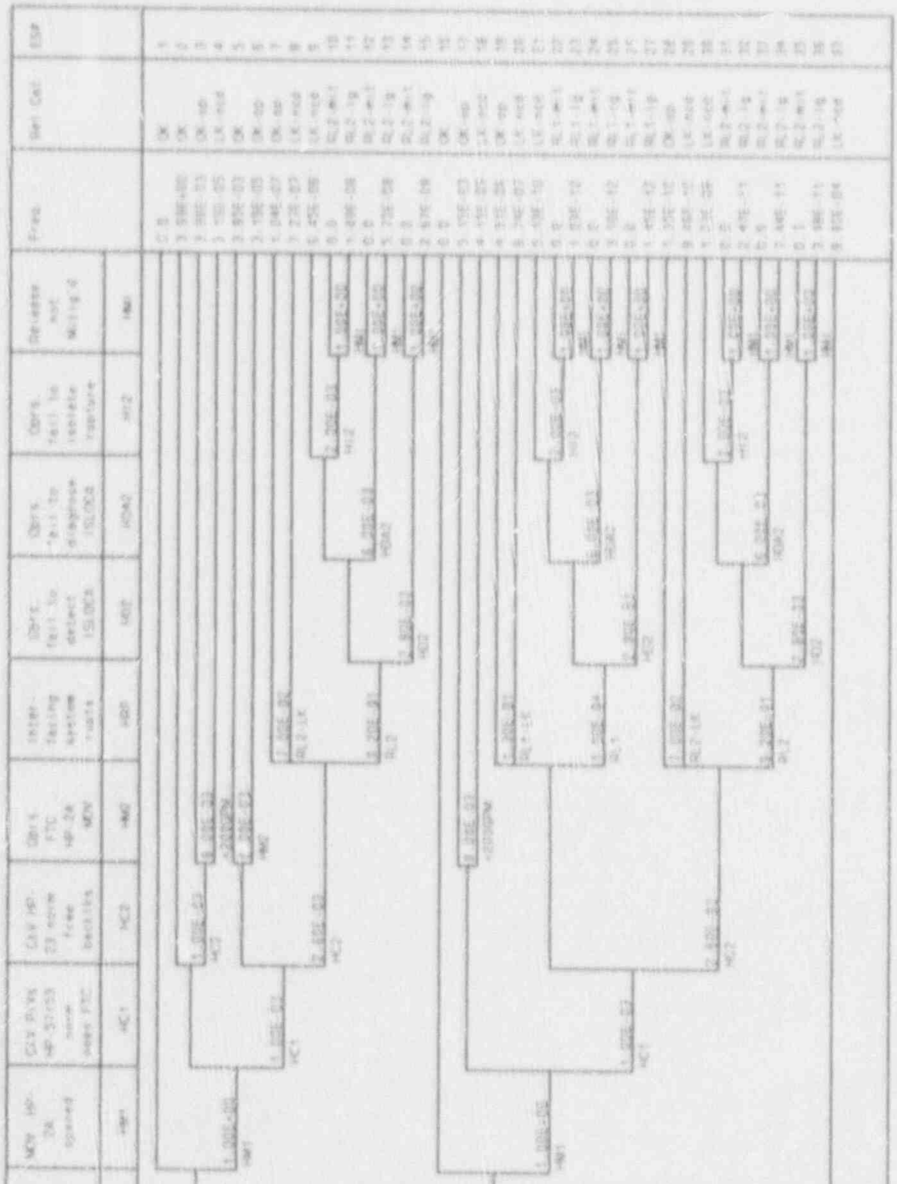

Figure 10. Reference B&W plant MU&P system interface event tree.

pump suction being the most likely location (at 71% probability, see Listing 18 in Appendix H). The MU&P event tree is described in Appendix D.

**4.3.2 High-Pressure Injection System Interface Event Tree—HPI.** Figure 11 shows a schematic diagram of the interface between the HPI system and the RCS. The ISLOCA event tree for this system is shown in Figure 12. Each of the two HPI pump trains branch into two injection legs, with each injection leg discharging into one of the RCS cold legs. As mentioned in the description of the MU&P event tree, the PIB is maintained by two check valves that are welded together, a normally closed MOV (stroke tested quarterly), and the HPI pump discharge check valve. Because the MU&P system provides normal makeup to the RCS through a connection in HPI leg A, that line is analyzed separately. The other three injection legs are modeled together in the HPI event tree.

The analysis of the HPI interface produced rupture scenarios that are identical to those discussed previously (for the HU&P event tree). The HPI event tree is described in more detail in Appendix D.

**4.3.3 DHR Letdown (Shutdown) Interface Event Tree—DHR-SD.** Once plant shutdown has been initiated, the control room operators monitor the primary system pressure and temperature in order to ensure adherence to the limits and requirements governing shutdown (e.g., at the reference B&W plant, the cooldown rate is limited to 100°F/hour for temperatures above 270°F, and 50°F for temperatures below 270°F). When the RCS temperature and pressure are reduced to approximately 280°F and 266 psig, respectively, DHR operation is initiated. Figure 13 shows a schematic diagram of the interface between the DHR letdown line and the RCS. The ISLOCA event tree for this interface is shown in Figure 14. The scenario of interest here begins with the premature opening of the DHR letdown line (MOVs DH-11 and DH-12) and is based on the premise that shutdown cooling has begun as a result of a human error. This generic error is the result of the control room operators either to misjudge the need for DHR, misread the cooldown curve, misinterpret the system indicators, or misunderstand the procedures and instructions, etc. The pressure and temperature of the RCS can be anywhere from 2,200 psi and 600°F to 266 psi and 280°F. The lower end of the pressure range would seem likely in those cases where plant shutdown proceeds expeditiously, while the high end of the range might be possible if the plant has spent an unusually long time in hot standby.

One area of interest relates to the plant procedures for initiating DHR operations. The two DHR letdown MOVs (DH-11 and DH-12) are interlocked with RCS pressure such that they cannot be opened if the RCS pressure is above 301 psi for DH-11 and 266 psi for DH-12. Because of a deadband in the interlock for DH-12, the procedure allows the operators to jumper-out the relays in order to bypass the interlock if the valve does not open. The contribution to the ISLOCA risk may be increased by a crew that is familiar with this portion of the procedure that allows bypassing this protective safety feature.

The key event for this sequence is the postulated human error of entering DHR cooling when it is not required. A human error of this type constitutes an error of commission, either in execution or intention. Typically, errors of commission have not been addressed in past probabilistic safety analyses. However, the existence of errors of commission is clearly supported by historical experience (i.e., LERs), and Appendix A contains a set of events in which errors of commission played a significant role. As discussed in Section 2, the LER data base was examined for events in which operators inappropriately opened MOVs. The result of this analysis, documented in Appendix A, produced error rates on the order of 1.0E − 07 per hour. Assuming a fault exposure time of about 10 hours per reactor-year (time associated with plant startup and shutdown when errors of this type are most likely), generate failure rates of 1.0E − 06 per reactor-year. This rate probably represents a lower bound on the industry wide average given the nonconservatisms associated with the LER system (see Appendix A

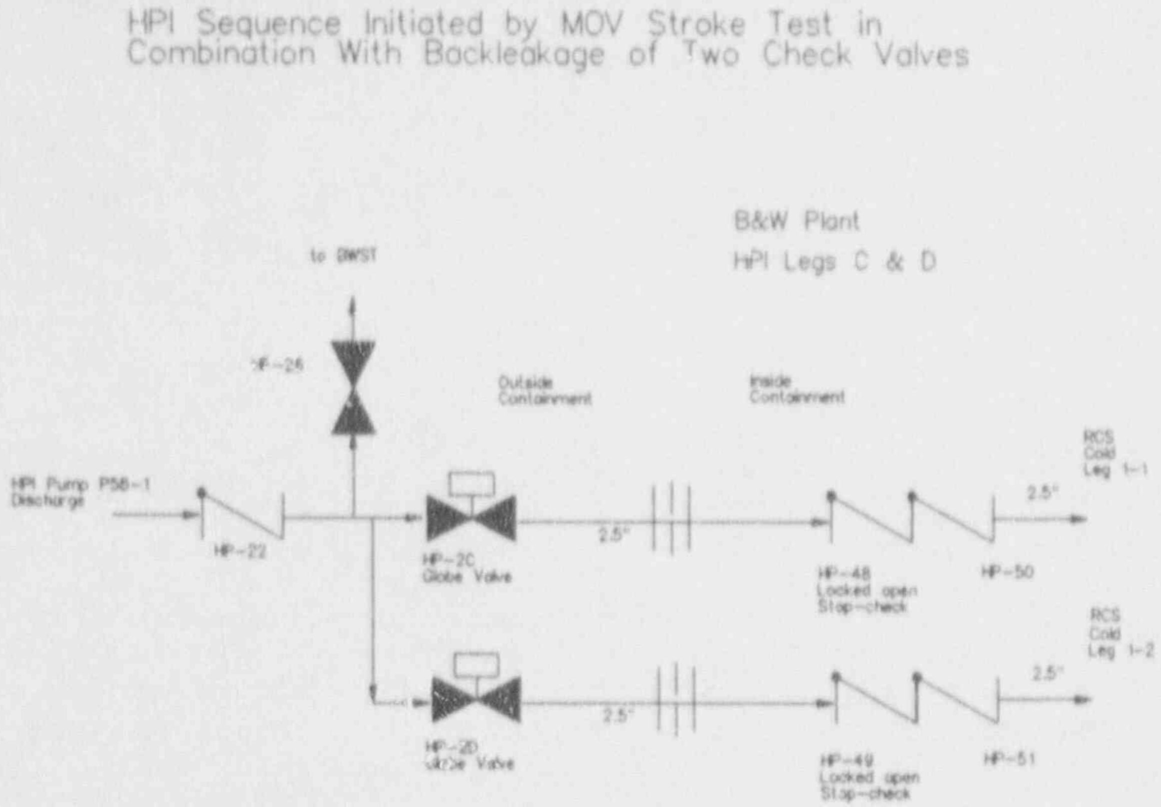


**Figure 11.** Schematic diagram of the HPI interface.

for a more complete discussion of using LERs for failure rates).

Assuming that it is possible for a human error to occur that allows for a premature initiation of DHR cooling, the probability of premature initiation is then a function of RCS pressure (i.e., an operations crew is more likely to enter DHR cooling at 400 psi in contrast to 2,000 psi). As a result, an exponential probability distribution was assumed (see Appendix D for a more complete description of this model). This distribution was used to weight the HEP by the expected RCS pressure. This modified HEP in turn was used to calculate the DHR system rupture probability. This process generated an aggregated large-rupture failure probability of 0.11 for the three likely large-rupture locations (see Appendix H, Listing 34, for pressure dependent system failure probabilities). These rupture locations can be identified from Listing 29 in Appendix H and are as follows: (a) the DHR heat exchanger, (b) the containment sump suction line between valves DH-2733(4) and DH-9B(A), and (c) the DHR letdown line immediately downstream of DH-11.

A bounding calculation was performed to estimate the time to core uncovery for this sequence. The estimate produced a minimum time of approximately 2 hours, which was used in the HRA to estimate HEPs for recovering from a DHR-ISLOCA sequence (see Appendix G). The event tree for this sequence is more fully detailed in Appendix D.

**4.3.4 DHR Letdown (Startup) Interface Event Tree—DHR-SU.** The DHR system may be overpressurized if the DHR letdown line remains open while the RCS is being heated up and pressurized. The schematic diagram of the DHR interface with the RCS is shown again in Figure 15 and the ISLOCA event tree for this system is shown in Figure 16. There are two ways in which RCS water can enter the DHR system:

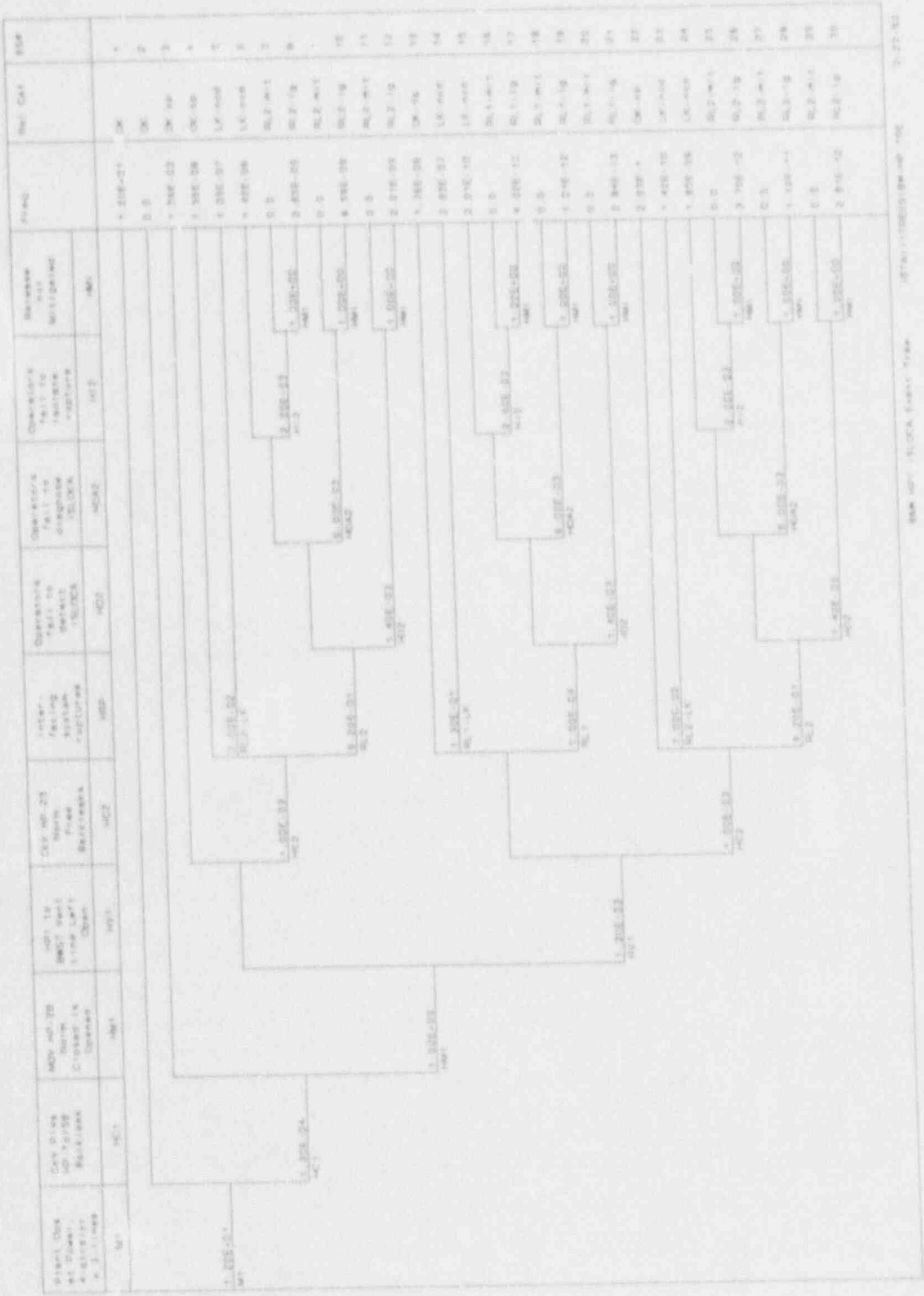

**Figure 12.** HPI ISLOCA sequence event tree.

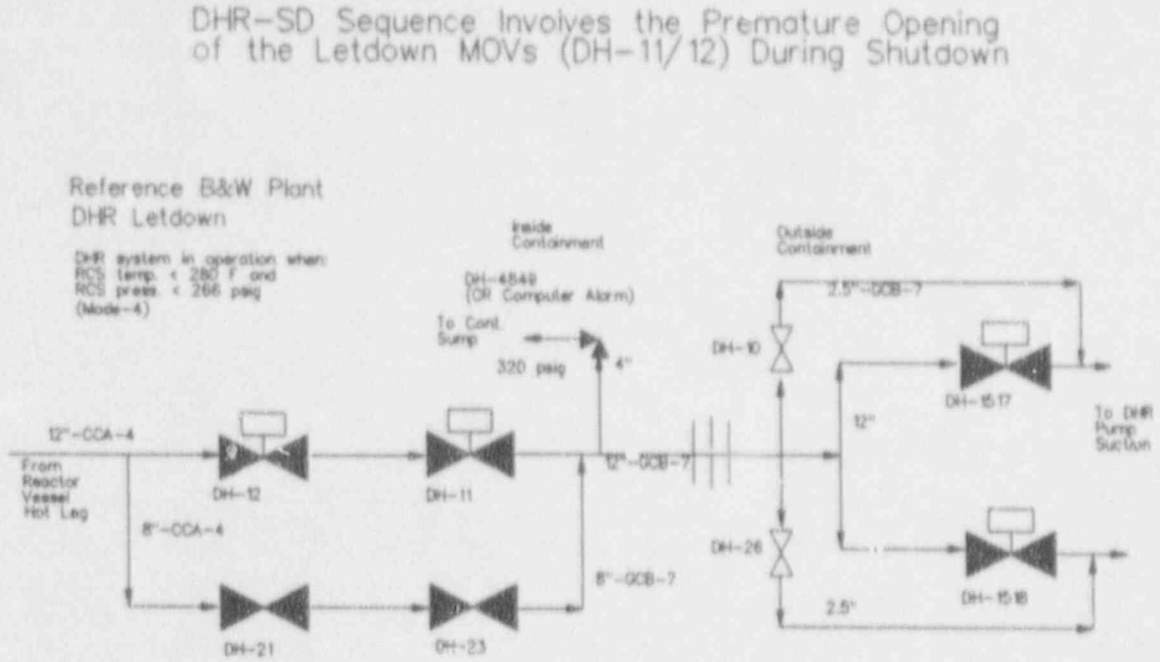


Figure 13. Schematic diagram of the DHR letdown interface (shutdown).

(a) via the normal letdown MOVs DH-11 and DH-12 and (b) via the MOV bypass valves DH-21 and DH-23, which are local-manually operated valves. DH-11 and 12 are interlocked to automatically close when the RCS pressure is above 300 psig. However, the valves always have their control power removed to prevent inadvertent operation, thus defeating the closure interlock. This is specified in the plant's Technical Specifications.

The DHR letdown line includes a pressure relief valve (DH-4849). This relief valve is important when analyzing the behavior of the RCS when the overall system is slowly pressurized with the letdown line left open. The makeup system pumps are each rated at 150 gpm and the relief valve (DH-4849) is designed to pass 1,800 gpm at 320 psi. Therefore, it does not seem possible to slowly pressurize the overall RCS to the point of rupturing the DHR, given the successful operation of DH-4849. The possibility of DH-4849 failing to open was also considered.

Other aspects of this sequence are similar to the DHR-SD sequence with respect to probable failure locations and time to core uncovery, for those scenarios where ruptures are possible. The event tree for this sequence is further described in Appendix D.

**4.3.5 Low-Pressure Injection System Interface Event Tree—LPI.** A schematic diagram of the LPI interface with the RCS is shown in Figure 17. The ISLOCA event tree for this system is shown in Figure 18. This interface represents the classical V-sequence configuration of two check valves in series, forming the PIB between the RCS and LPI system. The system comprises two redundant trains, with each injection line supplied by one LPI pump and one CFT. Pressure isolation valve failures have been investigated and it has been noted that pressure isolation check valves on CFT discharge lines have experienced a higher failure rate than other check valves[3] (note that this applies to check valves in standby service).

This sequence has two possible scenarios, depending on which check valves are postulated to fail. In the first scenario, CF-30(31) and CF-28(29) fail. This results in reactor coolant back-leaking into the CFT, which if ruptured, produces a loss-of-coolant accident (LOCA) inside containment. The second scenario pairs

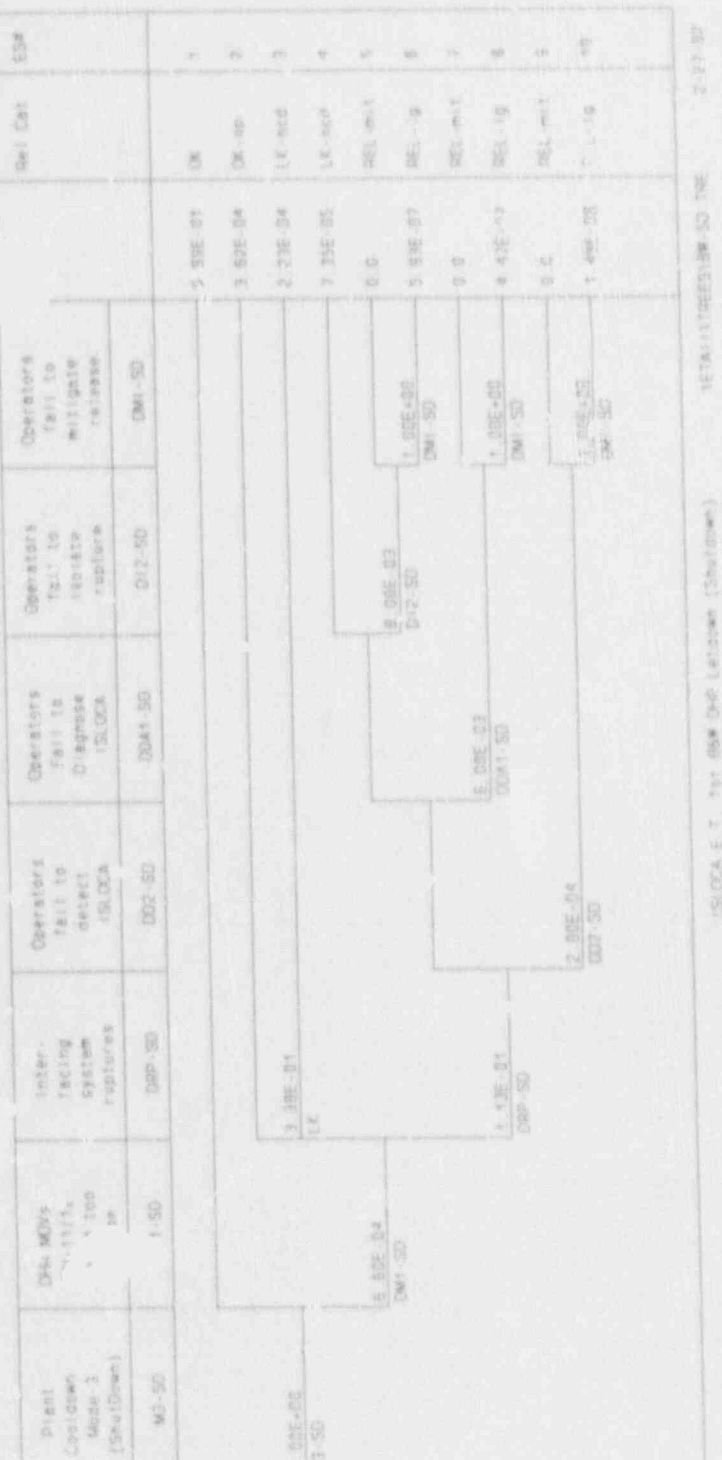

Figure 14. DHR letdown system (shutdown) ISLOCA sequence event tree.

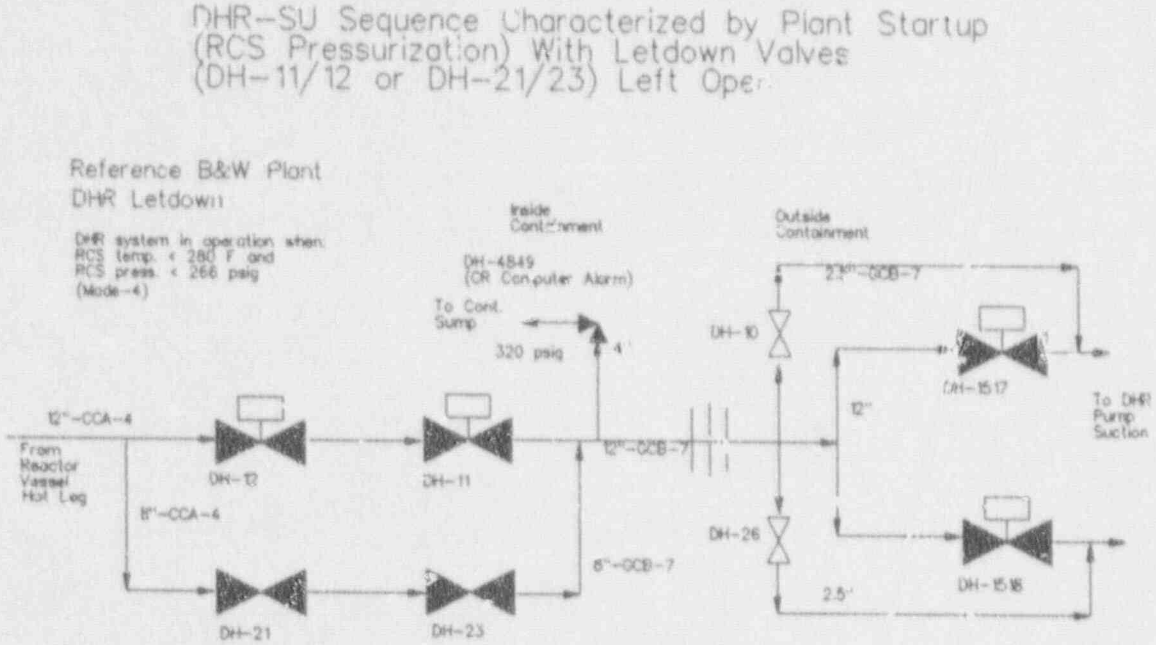


**Figure 15.** Schematic diagram of the DHR letdown interface (startup).

DH-76(77) with CF-30(31). In this case, the LPI system will be pressurized. Based on the results of the DHR analysis, the most likely failure locations will be the DHR heat exchangers. As with the DHR sequences, the time to core uncovery is estimated at approximately 2 hours. This event tree is more fully described in Appendix D.

**4.3.6 Auxiliary Building Environmental Analysis.** An important issue was developed during the course of the ISLOCA Research Program. This issue concerned the effects that an interfacing systems break would have on equipment in the break compartment and in adjacent compartments of the auxiliary building. This concern related to the possibility that redundant trains of emergency core cooling could be disabled by (a) high temperature, (b) high humidity, or (c) flooding. Thus, the assumption that only equipment in the break compartment is impaired by the break needs to be addressed in more detail.

Mechanistic calculations have been performed and analyzed to resolve the above issues. These calculations were performed for the reference B&W plant. The calculations provide a best estimate of the environmental conditions in the auxiliary building during an ISLOCA accident sequence. The parameters of interest are (a) pressure, (b) temperature, (c) relative humidity, and (d) water level in the affected ECCS equipment rooms. The calculations were used to estimate the failure potential of equipment in compartments adjacent to the break due to high temperature or submergence. Also the calculations were used to estimate the extent to which operator recovery actions would be limited by steam and water propagation through the auxiliary building.

Five break sequences were evaluated by using a combination of RELAP5 and CONTAIN thermal-hydraulic models.[21,22] The five sequences involved breaks in the following locations within the B&W reference plant's DHR/LPI and HPI systems:

1. **A 12-in. break with discharge into Room 236 of the auxiliary building.** This failure occurred as a result of premature entry into DHR cooldown, with the RCS at an elevated pressure (large break).

2. **A simultaneous rupture in both DHR heat exchangers, with discharge into Room 113 of the auxiliary building.** The limiting flow area for this break was in the 2.5-in. bypass lines around valves

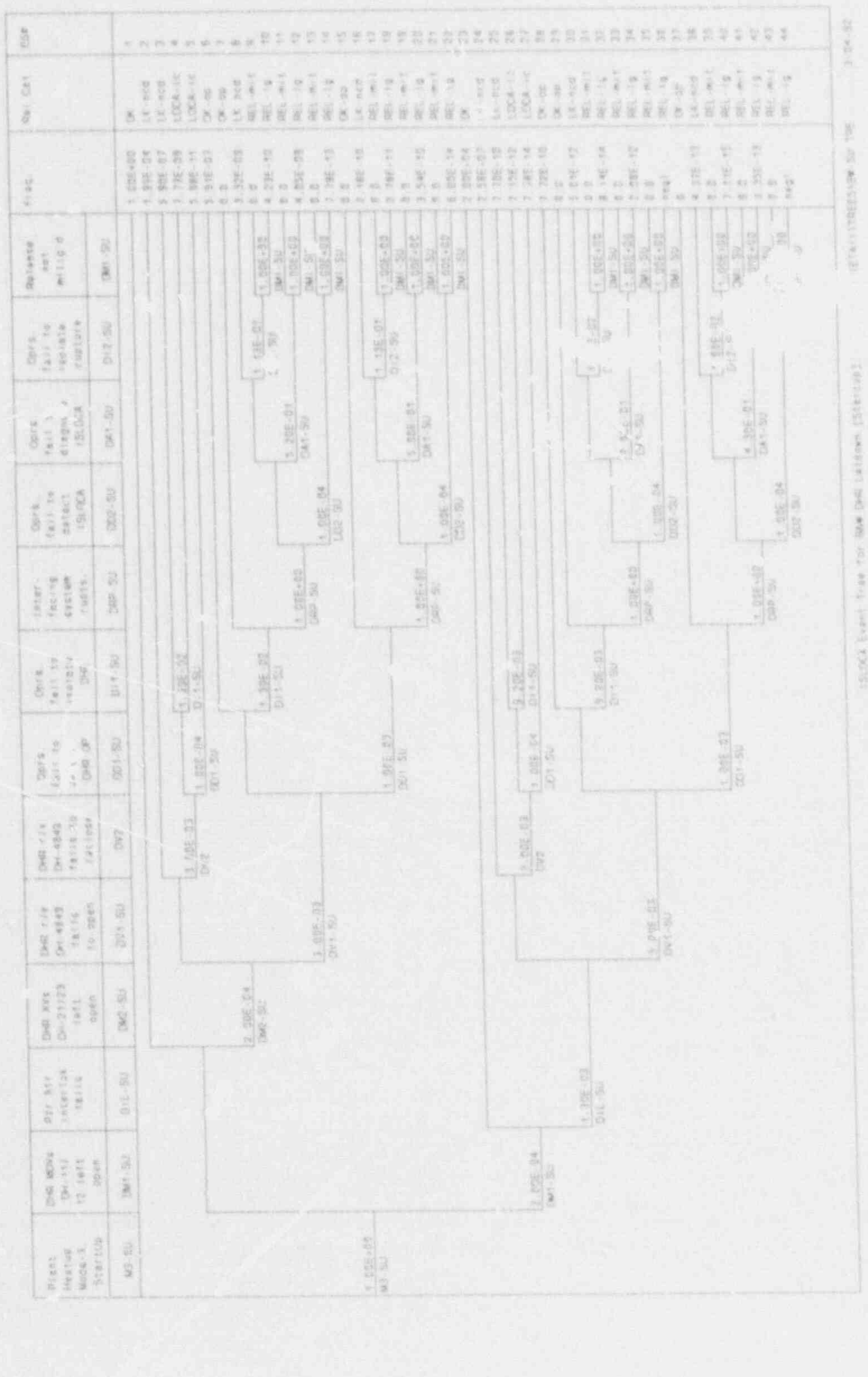

Figure 16. DHR letdown (startup) ISLOCA sequence event tree.

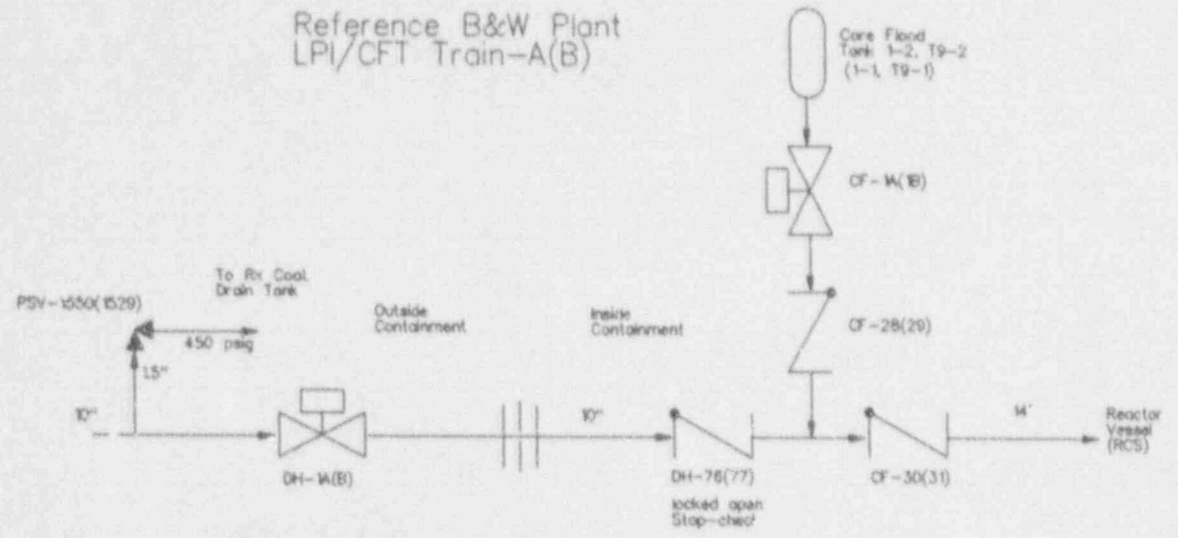


**Figure 17.** Schematic diagram of the LPI interface.

DH-1517 and DH-1518. This postulated failure occurs as a result of premature entry into DHR cooldown, with the RCS at an elevated pressure (large break).

3. **A simultaneous rupture in the low-pressure DHR pump suction piping, discharging into two rooms (105 and 113) of the auxiliary building.** The limiting flow area was in the 2.5-in. bypass lines around valves DH-1517 and DH-1518. This postulated failure occurs as a result of premature entry into DHR cooldown, with the RCS at an elevated pressure (small break).

4. **A rupture of the 1-2 decay heat cooler with a discharge into Room 113 of the auxiliary building.** The postulated break occurs because pressure isolation check valves in the injection piping of the LPI system failed internally (small break).

5. **A rupture in the suction piping to HPI pump 1-2 with discharge into Room 115 of the auxiliary building.** The postulated break occurs as a result of internal failure of the HPI discharge isolation check valves (small break).

The auxiliary building pressures, temperatures, and water levels depend on several variables. These variables are (a) the steam flow rate into the building, (b) the steam energy, (c) the volume of the auxiliary building, (d) the flow paths of the building, (e) the rate of heat removal by condensation of steam on structural materials and equipment, and (f) the effects of fire sprays. The steam energy and flow rate into the building were calculated with RELAP5[21] models of the RCS and the interfacing systems. The resulting steam source data were then used as boundary conditions for a CONTAIN[22] calculation of the auxiliary building response to the ISLOCA.

The RELAP5[21] models of the RCS comprised five volumes: (a) the cold leg, (b) lower plenum, (c) core, (d) upper plenum, and (e) the hot leg. Decay heat was modeled with a best estimate Oconee core model normalized to the B&W reference plant's operating power level. The ECCS injection flow was included in the model via pressure dependent flow tables. The pressure losses between the RCS and the break location were established by including a detailed model of the piping run. The purpose of this approach was to obtain an approximation of the RCS behavior. The modeling approach was based on the premise that the auxiliary building's response to an ISLOCA is not strongly dependent on the details of the RCS behavior.

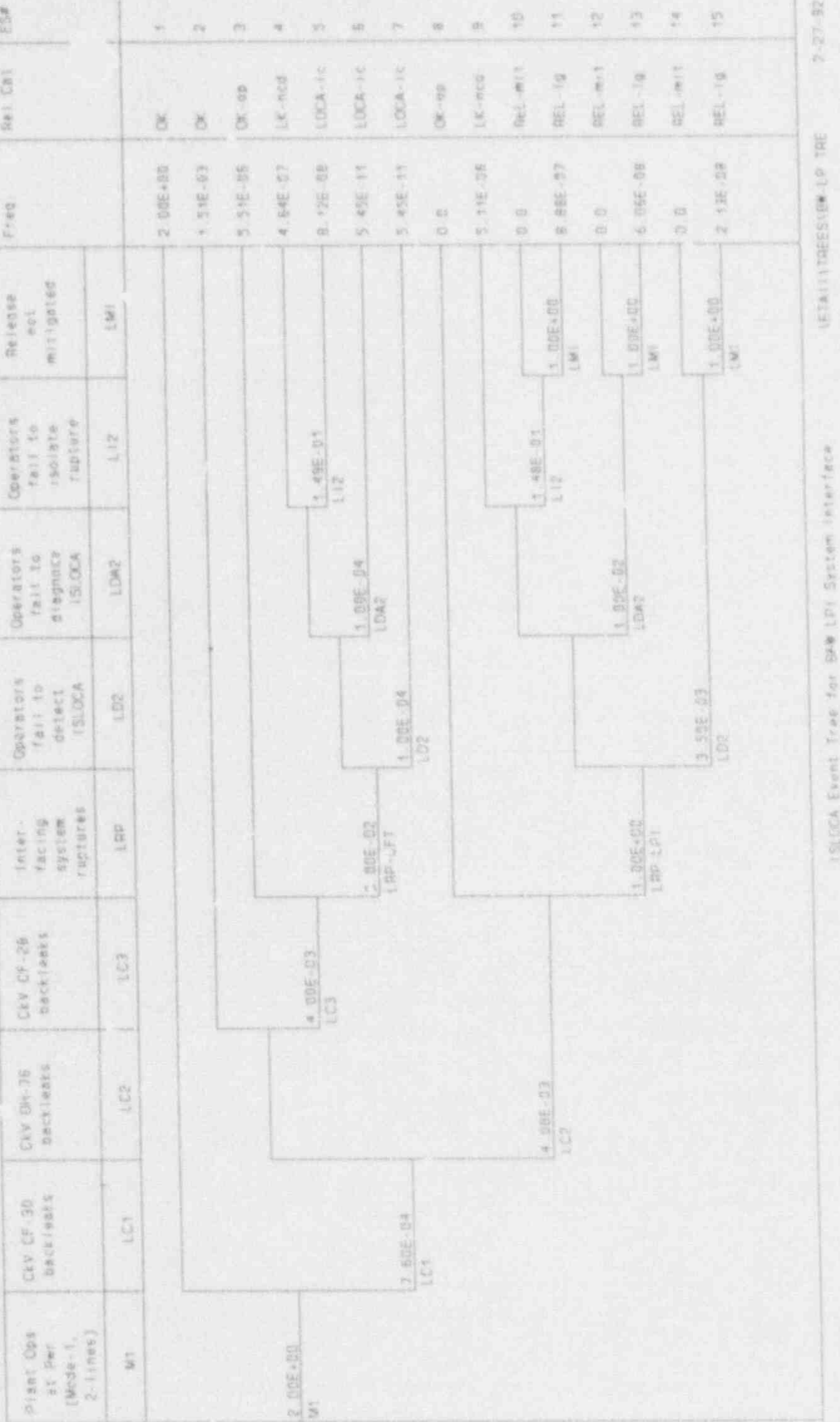


Figure 18. LPI ISLOCA sequence event tree.

The CONTAIN[22] computer code was used to calculate the time response of the auxiliary building parameters. The CONTAIN code is a containment modeling code. This software package has adequate steam condensation heat transfer models for the scenarios of interest. The code also provides engineered systems models. These system models allow analysis of fire sprays, compartment sumps, and drainage paths between compartments. The approach used in these calculations was to construct detailed cell models of compartments with ECCS equipment. These compartments were selected based on the belief that they might be affected by an ISLOCA. These rooms were identified to be auxiliary building rooms numbered in this analysis as 105, 113, 115, and 236. The remaining spaces in the plant were lumped into a single balance of plant volume.

The CONTAIN nodalization included a best estimate representation of the important geometric parameters that influenced the analysis. These parameters are the (a) gas volume of each compartment, (b) flow loss characteristics, (c) area of flow paths between each compartment, (d) description of the heat transfer surfaces within each compartment, and (e) description of drainage paths between compartments. The nodalization allowed the software to estimate pool depths in each compartment. In addition to liquid flow out the break, these water pools resulted from the condensation of steam, discharge from the fire sprays, and flow from adjacent compartments (both condensate and fire water). The flooding phenomena is an important aspect of the large break sequences. Flooding is important because the large break discharge flow rates quickly fill the compartment sumps. This discharge causes flooding of adjacent compartments and also results in flooding of compartments at lower elevations. One of the four compartments modeled in detail (Room 236) has fire sprays. These sprays can be expected to be actuated early in an ISLOCA sequence. The result is that the fire sprays can contribute to compartment and auxiliary building flooding.

Compartment sump pumps, fan coolers, and pump heat were excluded from the model. The sump pumps were excluded because they do not have adequate capacity to remove the break discharge condensate. Neither do the sump pumps have the capacity to remove the water discharged from the multiple firewater spray locations in the auxiliary building. The impact of sump pump operation was evaluated qualitatively in the small break sequences where the combined accumulation of water from these sources was close to sump pump capacity. Pump heat was neglected in the analysis because the heat would be removed by the fan coolers. The fan coolers were not modeled because a large fraction of their capacity would be used to remove pump heat.

The CONTAIN calculations performed for this B&W reference plant indicate that pressurization of the auxiliary building is limited to less than 1 psig. This is because the auxiliary building's compartments are well connected to one another. The modeling parameters that affect the calculated pressure rise between compartments are (a) flow area, (b) discharge coefficient, and (c) the length/diameter ratio. These parameters were all accurately modeled for the ECCS rooms.

The uncertainties associated with the pressure calculation are associated with the number of flow junctions neglected in the balance of plant portion of the model. The largest pressure drop in the auxiliary building's model occurs as the fluid passes from the break compartment to the immediately adjacent compartment. At each successive flow junction, the mass flow is reduced by the steam mass condensed by passing through each compartment. The effective flow area associated with the flow increases as more and more parallel flow paths become available to the fluid in the auxiliary building. Pressure drops in the auxiliary building at each succeeding junction therefore decrease as distance from the break increases. This pressure drop behavior of the auxiliary building leads to an important conclusion. This conclusion is that refinements to the CONTAIN nodalization would not have a appreciable affect on the calculated peak pressure rise of the auxiliary building.

The CONTAIN software package predicts that for the range of break sizes analyzed, the temperatures in the auxiliary building do not exceed 212°F.

There is one modeling uncertainty that can change this result significantly. This uncertainty is associated with the quality of the water-steam mixture discharged from the RCS break into the auxiliary building. For the break sequences analyzed, the break discharge is a two-phase mixture with a steam quality no higher than 0.90. Because CONTAIN models the RCS blowdown flow as an isenthalpic expansion from the RCS pressure to compartment pressure, the resulting compartment temperatures will always be at, or very near, the saturation temperature associated with the calculated compartment pressure. This is true as long as the break quality does not exceed approximately 0.93. At break qualities higher than 0.93, the enthalpy of the discharge fluid will be high enough to produce superheated steam in the compartment. The maximum steam temperature obtainable by this process is approximately 320°F. This temperature occurs when dry saturated steam at approximately 500 psia is discharged from the RCS through the break. It must be emphasized that none of the RELAP5 RCS model predictions indicate that dry steam will be present in the break discharge. Given the uncertainty inherent in calculations of this type, the RELAP5 calculations cannot completely rule out the possibility that high-quality steam will be discharged during an ISLOCA for sufficient time to superheat the steam in the break compartment. All best-estimate calculations completed to date indicate that the break's compartment does not superheat, and this lack of superheating was used in the analysis of the ISLOCA events.

The relative humidity of the auxiliary building was analyzed from the CONTAIN calculations. The CONTAIN relative humidity predictions were similar in each sequence. All of the auxiliary building rooms that were modeled experienced extended periods in which the relative humidity was 100%.

The rate of flooding varied considerably among the five sequences. In the DHR sequence in which the large break occurs in the letdown line, the source of the flooding is located in Room 236. This flood propagated through a pipe chase in the floor of Room 236 into Room 115. The flood then submerged all the ECCS pump motors in a little over 30 minutes. This was primarily the result of the propagation of the unflashed portion of the break discharge. In this sequence, the operation of the sump pump would have been ineffective in mitigating the flooding. This inability of the sump pumps to mitigate the flooding is a result of the low pumping capacity of the sump pumps.

The pool depths and resulting flooding associated with the small break sequences was minimal. The flooding was minimal because the mass discharged from the break was relatively small and most of the discharge flashed to steam. The steam was then carried through the auxiliary building. The discharged steam was either released to the atmosphere through blow out panels or was condensed away from the break's location. The sump pumps, if they had been included in the model, would have slowed the pool growth and delayed the submergence threat to the ECCS equipment. The flooding, associated with the small break ISLOCA sequences, occurs primarily because firewater discharged from the sprinkler system in Room 236 drains into the ECCS compartment below (Room 115). At the end of 2 hours, the flood had not yet reached the top of the spill wall separating Room 115 and Room 113. Equipment in Room 105 would not be threatened for a number of hours.

The flooding results can be summarized with two general statements:

- For large break sequences, flooding will occur in the break compartment and in adjacent compartment at a rate that will cover essential ECCS components within one hour
- For small break sequences, flooding will occur slowly and could be delayed by the operation of the compartment sump pumps. A period of many hours would pass before essential ECCS components would be threatened.

Of the various factors controlling pool formation, two dominate. The first is the rate of discharge of unflashed fluid from the break. The second is the extent to which firewater and condensate from the balance of plant find their way into rooms housing ECCS equipment. For the large break sequences, the principle contributor to pool formation is the discharge liquid that does not flash to steam. For large break sequences as the RCS cools down, this becomes essentially the run out flow of the surviving ECCS. In the small break sequences, the discharge of the fire protection sprays provides a greater flooding hazard than the accumulation of condensate or unflashed break discharge.

The CONTAIN results from the steam propagation analysis show that the operation's crew entry into the auxiliary building would be prohibited by the live steam environment. This steam environment forms within a few minutes of system rupture. The CONTAIN results also indicate that the most limiting environmental factor is water pool formation (flooding) in the ECCS compartments. When the pools reach a depth of 2 ft, the ECCS pump motors will become submerged. This submergence will cause failure of the ECCS pumps. The time at which this failure occurs becomes the limiting time available for operator recovery. The temperature and humidity effects were not important at the B&W reference plant because all of the ECCS equipment was qualified for the postulated environment produced by a high-energy line break.

**4.3.7 ISLOCA Sequence Timing and Operator Actions.** Given a rupture has occurred, the likelihood is high that the operators will isolate the break and recover from the emergency situation. However, to adequately assess the probability of successfully recovering from an emergency situation, an analysis must be done that comprises a number of aspects. The first is the identification of those influences important to the probability of successful recovery. Among these are the time available for the operators to act, the availability and accessibility of equipment necessary for recovery, and the human-factor aspects such as indicators and procedures. Once the particular scenario and its context have been defined, an HRA can be performed to determine the probabilities of detecting the LOCA situation, diagnosing the rupture as being outside containment, and recovering from the emergency situation by isolating the break. This section will address these scenario specific aspects of this issue; specifically, the sequence timing, available equipment, and indications and procedures will be discussed.

***4.3.7.1 Timing to Core Uncovery.*** The time available for recovering from an accident sequence is defined as the time to core uncovery in this analysis. The time to core uncovery is assumed to be the point at which core damage begins to occur. Two independent sets of calculations were performed to estimate this time and are documented in Appendix G and Appendix M.

The two different core uncovery time calculations used in the analysis are based on two different controlling phenomena. The calculations in Appendix G are based on the water inventory available for injection into the primary system. For the sequences postulated in this analysis, the duration of emergency coolant injection is limited by the available water inventory because the LOCA occurs outside of the containment. In these accidents, the leaking coolant cannot be recirculated via the emergency sump located in the containment. Core uncovery follows once the ECCS water supply is depleted and coolant injection stops.

The calculations of Appendix M are based on the environmental effects that an interfacing system rupture potentially has on ECCS equipment survival in the auxiliary building. These calculations consider the possible consequences of discharging primary system coolant into the areas where vital equipment are housed. The calculations provide information on parameters that influence the environmental conditions that degrade the operation of the ECCS. These environmental effects include high temperatures, high humidities, and flooding. Direct spray and possible pipe whip are also considered. In this analysis, the assumption is made that all equipment in the immediate vicinity of the rupture fails at the time of the break. Severe environmental conditions are a concern in that they can propagate to adjacent compartments where redundant equipment is housed. For some rupture sequences, these environmental conditions can threaten to fail all the ECCS equipment. In all of the sequences examined, flooding is the primary concern. If a rupture occurs, the discharge accumulates in a single room or compartment. When the water level reaches a certain point (either the top of a flood wall or barrier, or the level at which the door is burst open), then the flooding propagates into an adjacent compartment. If the rupture is relatively large, the flooding might propagate two or three compartments removed from where the break occurs. This is particularly significant when the first area to be flooded disables one train of the ECCS. As the severe environmental conditions spread to other compartments, the remaining ECCS train is threatened. Core uncovery follows once the ECCS fails and the coolant injection stops. The areas of concern are listed below (545 ft elevation is the lowest level of the auxiliary building; Room 236 is directly over and connected to Room 115):

- Room 105—ECCS Pump Room 1-1, elevation 545 ft, fire area AB
- Room 113—DHR Heat Exchanger Pit, elevation 545 ft, fire area AB
- Room 115—ECCS Pump Room 1-2, elevation 545 ft, fire area A
- Room 236—No. 2 Mechanical Penetration Room (contains the DHR letdown line), elevation 565 ft, fire area A.

Tables 13 and 14 show the results of the two sets of calculations. These calculation estimate the core uncovery times for different ISLOCA sequences.

**Table 13.** Summary of ISLOCA times to core uncovery based on exhaustion of water inventory (from Appendix G).

| Basis for time estimate | 2.5-in. HPI ISLOCA | 10-in. LPI ISLOCA |
|---|---|---|
| Time to empty BWST (hour)[a] | 2.90 | 1.1 |
| Time to core uncovery (hour) | 4.0 | 1.9 |

a. All times referenced to the beginning of the ISLOCA.

**Table 14.** Summary of ISLOCA times to core uncovery based on time to fail all ECCS by flooding (from Appendix M).

| Break sequence (see Appendix M) | Failure of all ECCS (hour)[a] | Time to core uncovery (hour) |
|---|---|---|
| BS-1 (DHR letdown line) | 0.6 | 1.5 |
| BS-2 (DHR Hx) | 2 | 3 |
| BS-3 (DHR pump suction) | 3 | 4 |
| BS-4 (DHR Hx via LPI) | 1 | 2 |
| BS-5 (HPI pump suction) | 3 | 4 |

a. All times referenced to the beginning of the ISLOCA.

The HRA used only rough estimates of the time to core uncovery. This is judged adequate since even the shortest time estimate is sufficient for performing the necessary recovery actions. That is, time constraints do not adversely affect the performance of the operators and is therefore not a critical parameter in the analysis. Other factors contributing to the decision to use rough time estimates in the HRA are the uncertainties in the codes, the models, and boundary conditions of the ISLOCA. The core uncovery times used in the HRA for the B&W reference plant are 2 hours for the DHR and LPI sequences and 4 hours for the HPI and MU&P sequences.

***4.3.7.2 Equipment Available for Recovery.*** There are isolation valves that when closed will isolate the break in all of the ISLOCA scenarios studied here. This capability to isolate ISLOCA leaks is believed to be generic. As an example of the ability to isolate a pipe break, consider the following two recovery actions:

1. When the initiating event is a failure of a check valve, there are typically motor operated isolation valves available that could be closed to provide isolation.

2. When the primary interfacing system isolation is via MOVs, their dominant failure mode is inadvertent operation (caused by human error). Inadvertent operation of MOVs is usually recoverable.

One break isolation issue that does remain is the capability of the MOVs to operate in a high delta-P environment. For this analysis, calculations were made to estimate the maximum delta-P in which the valves could operate. These calculations are documented in Appendix C, Section C.5. The results of these calculations support the conclusion that the MOVs identified for the B&W reference plant have the capability of closing on high delta-P. These calculations are based on new INEL experimental data. The new data and the calculations of Appendix C are not in complete agreement with the valve's original design information.

***4.3.7.3 Indicators and Procedures.*** If an ISLOCA were to occur, instrumentation is such that there would be adequate indications to allow the operators to correctly diagnosis the situation. This statement is true provided the operators do not become distracted and pursue an inefficient or inappropriate course of action. The ISLOCA indicators the operations crew have to base their actions on include the primary system monitors that signal a loss-of-coolant; local-condition sensors on the interfacing systems such as temperature, pressure, and flow sensors; and auxiliary building alarms such as sump pump operation, radiation alarms, and possibly the fire detection systems.

The primary concern in ensuring proper break isolation is the applicability of the emergency operating procedures under ISLOCA conditions. The interfacing system accident sequences postulated in this analysis for the reference B&W plant would be addressed in the plant's alarm panel annunciator response procedures. Consequently, a considerable amount of time could pass before the control room crew systematically investigates the cause of the "LPI low flow" alarm. The alarm procedure instructs the control room crew to dispatch an operator to inspect any accessible LPI or DH system piping. The philosophy used in this analysis assumed if the operators recognized the nature of the situation (based on the available indications), they would then take the necessary corrective action without expending significant amounts of time.

## 4.4 ISLOCA CDF Estimation

Based on the event trees described in Section 4.1 (and in more detail in Appendix D), the total ISLOCA CDF for the B&W reference plant is estimated to be 2.2E−06/reactor-year. Table 15 and Figure 19 provide breakdowns of this frequency by sequence and release category. The dominant sequence is the premature opening of DH-11 and DH-12 during shutdown (identified as the DHR-SD sequence). The results show that this human error initiated sequence along with one other (MU&P) contribute slightly more to the CDF than the multiple passive hardware failure

Table 15. Plant-damage state frequencies from ISLOCA sequences for a B&W reference plant (frequency per reactor-year).

| Sequence | CDF | REL-lg | REL-mit | LOCA-ic | LK-ncd | OK-op |
|---|---|---|---|---|---|---|
| DHR-SU | 4.9E−09 | 4.9E−09 | 0.0E+00 | 7.8E−09 | 2.0E−04 | 5.9E−07 |
| DHR-SD | 1.1E−06 | 1.1E−06 | 0.0E+00 | 0.0E+00 | 3.0E−04 | 3.6E−04 |
| LPI | 9.7E−07 | 9.7E−07 | 0.0E+00 | 8.1E−08 | 5.6E−06 | 5.5E−06 |
| HPI | 1.4E−08 | 1.4E−08 | 0.0E+00 | 0.0E+00 | 1.8E−06 | 1.6E−03 |
| MU&P | 1.0E−07 | 1.0E−07 | 0.0E+00 | 0.0E+00 | 9.7E−04 | 9.1E−03 |
| Totals | 2.2E−06 | 2.2E−06 | 0.0E+00 | 8.9E−08 | 1.5E−03 | 1.1E−02 |

Total CDF: 2.2E−06/reactor-year (sum of large and mitigated release frequencies).

Plant-damage state definitions:

REL-lg–Core damage with a large unmitigated radioactive release.
REL-mit–Core damage but radioactive release is mitigated.
LOCA-ic–Sequence results in a LOCA inside containment.
Leak-ncd–Reactor coolant is lost, but is either too small to be significant or is isolated before core damage occurs (no core damage).
OK-op–Interfacing system is overpressurized but does not rupture.

sequence (i.e., LPI). The hardware dominated sequences are similar to the classical event-V category of sequences that are typically examined in current PRAs. One consequence of modeling human error initiated sequences is the relatively high likelihood of leaks and overpressure events predicted by the models. This finding is supported by historical experience, which indicates that improper valve lineups are much more likely causes of operational events than are hardware failures. Further, the historical experience indicates that sudden, catastrophic valve failure has never occurred. Consequently, although less conservative hardware failure probabilities were used in this study than are often used in PRAs, the inclusion of human error contributions to the ISLOCA sequences produces an overall ISLOCA CDF slightly higher than those typically reported in past PRAs.

## 4.5 Risk Assessment

The consequences of the core damage producing ISLOCA sequences were estimated using the modified containment bypass source term from the Oconee PRA[7] and estimated DFs based on NUREG-1150.[2] Although the Oconee results were developed some years ago, the current generation of source term estimates are not significantly different. After the conditional consequences were calculated using the modified Oconee information and the estimated DFs, they were combined with the release category frequencies calculated here producing a point estimate of the ISLOCA risk. (The conditional consequences for a range of decontamination factors are listed in Table 16.) Two release categories were used for binning the event tree end states: mitigated and unmitigated (i.e., large) releases. A DF of 1 was assumed for the auxiliary building release (large or dry release) and a DF of 10 was used for the mitigated release (fire protection sprays or wet release). NUREG-1150 used a weighted average DF for the wet release of 18.5 and a median value of 8.5.[26] Additional work on estimating DFs for auxiliary buildings has been sponsored by Electrical Power Research Institute (EPRI) using the Modular Accident Analysis Program (MAAP) 3.0B code.[a] This work would seem to support DFs

a. Electric Power Research Institute, *Evaluation of the Consequences of Containment Bypass Scenarios*, EPRI-NP-6586-L, November 1989. This report contains proprietary information that is not available to the general public; however, the results of this study were made available to the INEL analysts for review.

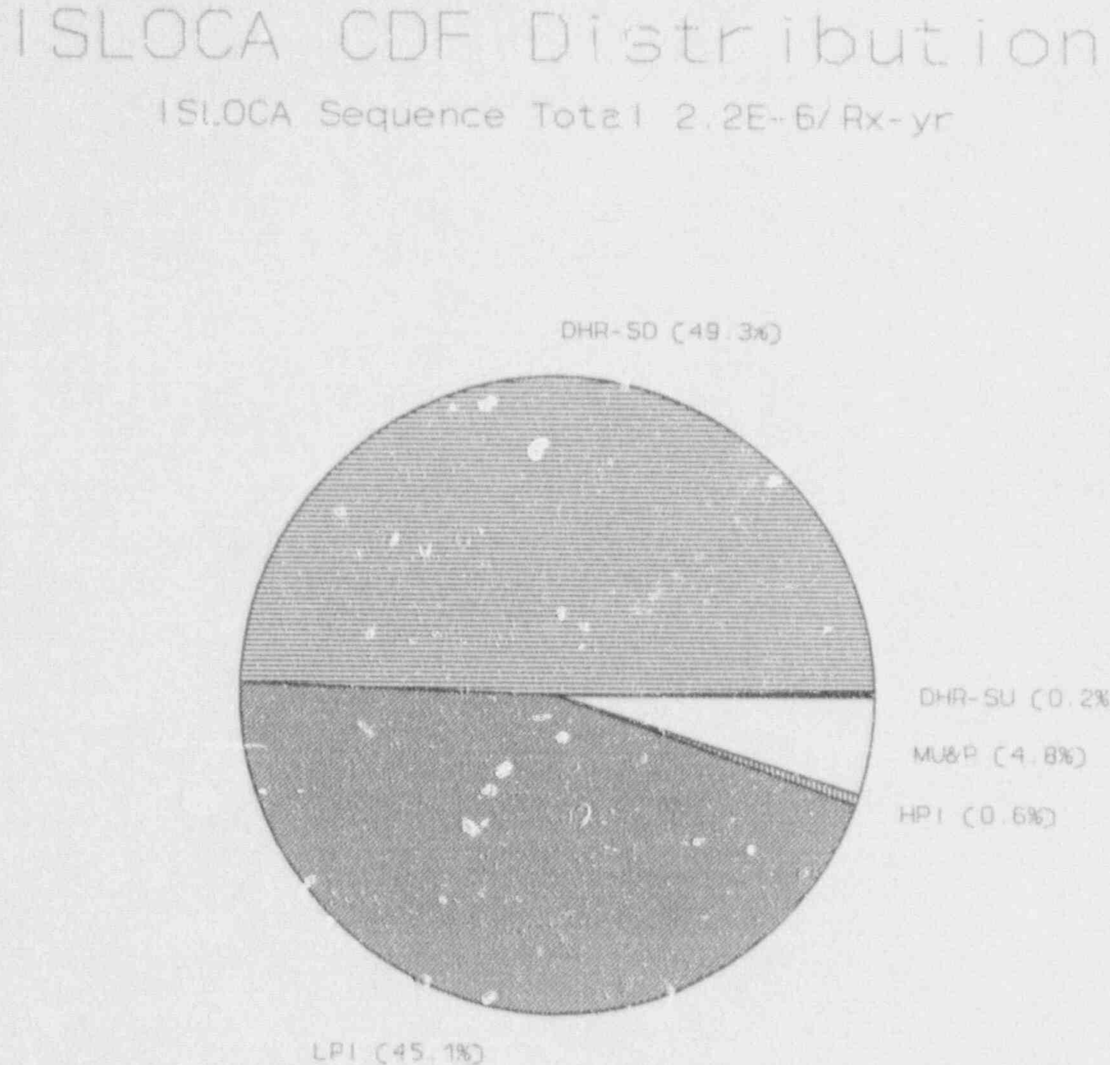


**Figure 19.** Sequence contribution to total ISLOCA CDF.

**Table 16.** MACCS consequence results for a range of possible DFs (Oconee source term, scaled to the B&W reference-plant power, and the Surry site).

| Consequence measure | DF = 1 | DF = 5 | DF = 10 | DF = 100 |
|---|---|---|---|---|
| Population dose (person-rem, 50-mi) | 2.8E+06 | 1.3E+06 | 9.7E+05 | 2.9E+05 |
| Latent cancers (total grid) | 4.5E+03 | 1.5E+03 | 8.9E+02 | 1.4E+02 |
| Early fatalities | 3.6E-02 | 3.0E-04 | 5.8E-05 | 1.2E-06 |

for a dry release in the range of 3 to 80, depending on the specific configuration of the auxiliary building. Wet release DFs, with either a flooded break location or sprays available, ranged from 40 on up. These DFs while not fully supported by experimental data, tend to indicate that the source terms used in this analysis provide a reasonable estimate of the expected consequences of an ISLOCA. In addition, the EPRI calculations may have under-estimated the amount of hydrogen expected to be generated. Increasing the amount of hydrogen could (if the hydrogen burns) lead to reduced DFs.

When reviewing the consequence and risk estimates, several aspects of this calculation should be kept in mind. Many measures of risk are available and have been used in recent studies (e.g., NUREG-1150[2]). However, to produce these estimates, many sequence-specific and site-specific assumptions must be made from the cost of land to the warning time available before a release occurs. These assumptions can have significant effects on the consequences calculated by MACCS.[20] For example, a sensitivity calculation was made (although not documented) where the release timing was changed from that postulated for the Oconee V-sequence to that assumed for the original NUREG-1150 Surry analysis. Population dose and cancer fatalities were relatively unchanged, but early fatalities increased by a factor of 20. The major influence on this change is the relation between the start of evacuation (determined by summing the warning time and the evacuation delay time) and the time and duration of the release. Conversely, population dose and latent cancers, which are the result of long-term effects (i.e., the people move back onto slightly contaminated land and accumulate a lifetime radiation dose), seem to be very robust consequence measures, insensitive to timing assumptions. The ISLOCA risks for the reference B&W plant are shown in Table 17.

## 4.6 Sensitivity Study Results

Because human errors dominate the results for the reference B&W plant, the major effort in evaluating the effects of uncertainty, and the sensitivity of major issues on risk, was devoted to the HRA. The one exception to this is an analysis of the effects of the uncertainty in pipe rupture pressures on the CDF of the DHR-SD sequence, which will be described first.

**4.6.1 Pipe Rupture Pressure Uncertainty.** Because of the difficulty in predicting the existence of flaws in piping, the uncertainty on the median pipe rupture pressure is estimated to be greater than for any other component. The relatively wide distribution on failure pressure generates correspondingly large failure probabilities for system pressures less than the median failure pressure. It was therefore decided to test a hypotheses that the relative importance of pipe ruptures in ISLOCA sequences was an artifact of this uncertainty. The DHR-SD sequence was chosen for evaluating the effect of the uncertainty in pipe rupture pressures on the CDF for the following two reasons: (a) it represents the dominant core damage sequence, and (b) it is analyzed on a weighted average of the range of possible system

**Table 17** ISLOCA risk for the B&W reference plant (Oconee source term, scaled to reference-plant power, and the Surry site).

| Risk measure | REL-lg DF = 1 | REL-mit DF = 10 | Total |
|---|---|---|---|
| Population dose (person-rem, 50-mi) | 6.0 | 0 | 6.0 |
| Latent cancers (total grid) | 9.6E − 03 | 0 | 9.6E − 03 |
| Early fatalities | 7.7E − 08 | 0 | 7.7E − 08 |

pressures. That is, system rupture calculations were performed for RCS pressures ranging from 400 to 2,200 psig. The conditional rupture probabilities were then weighted by the probability that the control room operators would prematurely open the DHR-letdown isolation valves (DH-11 and DH-12). The HEPs, in turn, are dependent on the RCS pressure such that it is about 1,000 times less likely that the valves would be prematurely opened at 2,200 psig when compared to the HEP at 400 psi. This process produced a large-rupture probability of 0.11, conditional on the premature opening of DH-11 and DH-12 at some unspecified RCS pressure between 400 and 2,200 psig.

The uncertainty in the pipe rupture pressure is expressed as the logarithmic standard deviation of the failure pressure. The failure pressure is assumed to be lognormally distributed. The logarithmic standard deviation then describes the spread in the corresponding normal distribution of the failure pressure [i.e., the standard deviation of ln(failure pressure)]. The best estimate of the logarithmic standard deviation is 0.36, which represents the base case. The sensitivity case was calculated assuming the uncertainty in the failure pressure could be reduced such that the logarithmic standard deviation (log-std-dev) would be 0.1. This represents an extremely low value for this parameter (subsequent information indicated that a value of 0.26 would be a realistic lower limit). Tables 18 and 19 show the pressure dependent and the HEP-weighted system rupture

**Table 18.** DHR system rupture probabilities (weighted by the HEP of prematurely opening DH-11/12) as a function of RCS pressure (pipe failure pressure log-std-dev = 0.36).

| RCS pressure (psig) | HEP | System rupture probability | | | HEP-weighted system rupture probability | | |
|---|---|---|---|---|---|---|---|
| | | Large | Small | No leak | Large | Small | No leak |
| 2,200 | 2.1E-07 | 1 | 0 | 0 | 2.1E-07 | 0 | 0 |
| 2,100 | 3.1E-07 | 9.99E-01 | 1E-03 | 0 | 3.1E-07 | 4.3E-10 | 0 |
| 2,000 | 4.5E-07 | 9.97E-01 | 3E-03 | 0 | 4.5E-07 | 1.1E-09 | 0 |
| 1,900 | 6.7E-07 | 9.95E-01 | 5E-03 | 0 | 6.6E-07 | 3.1E-09 | 0 |
| 1,800 | 9.8E-07 | 9.94E-01 | 6E-03 | 0 | 9.7E-07 | 6.3E-09 | 0 |
| 1,700 | 1.4E-06 | 9.91E-01 | 9E-03 | 0 | 1.4E-06 | 1.3E-08 | 0 |
| 1,600 | 2.1E-06 | 9.83E-01 | 1.7E-02 | 0 | 2.1E-06 | 3.7E-08 | 0 |
| 1,500 | 3.1E-06 | 9.64E-01 | 3.6E-02 | 0 | 3.0E-06 | 1.1E-07 | 0 |
| 1,400 | 4.5E-06 | 9.20E-01 | 8.0E-02 | 0 | 4.2E-06 | 3.6E-07 | 0 |
| 1,300 | 6.7E-06 | 8.36E-01 | 1.64E-01 | 0 | 5.6E-06 | 1.1E-06 | 0 |
| 1,200 | 9.8E-06 | 7.05E-01 | 2.95E-01 | 0 | 6.9E-06 | 2.9E-06 | 0 |
| 1,100 | 1.4E-05 | 5.51E-01 | 4.49E-01 | 0 | 7.9E-06 | 6.4E-06 | 0 |
| 1,000 | 2.1E-05 | 4.03E-01 | 5.97E-01 | 1E-04 | 8.5E-06 | 1.3E-05 | 2.1E-09 |
| 900 | 3.1E-05 | 2.81E-01 | 7.18E-01 | 1E-03 | 8.7E-06 | 2.2E-05 | 2.5E-08 |
| 800 | 4.5E-05 | 1.78E-01 | 8.10E-01 | [illegible]E-02 | 8.1E-06 | 3.7E-05 | 5.3E-07 |
| 700 | 6.7E-05 | 1.00E-01 | 8.09E-01 | 9.1E-02 | 6.7E-06 | 5.4E-05 | 6.1E-06 |
| 600 | 9.8E-05 | 5.0E-02 | 5.80E-01 | 3.70E-01 | 4.9E-06 | 5.7E-05 | 3.6E-05 |
| 500 | 1.4E-04 | 2.1E-02 | 1.93E-01 | 7.86E-01 | 3.1E-06 | 2.8E-05 | 1.1E-04 |
| 400 | 2.1E-04 | 7E-03 | 1.2E-02 | 9.81E-01 | 1.4E-06 | 2.6E-06 | 2.1E-04 |
| Total | 6.6E-04 | | | | 1.13E-01 | 3.38E-01 | 5.48E-01 |

**Table 19.** DHR system rupture probabilities (weighted to the HEP of prematurely opening DH-11/12) as a function of RCS pressure (pipe failure pressure log-std-dev = 0.1).

| RCS pressure (psig) | HEP | System rupture probability | | | HEP-weighted system rupture probability | | |
|---|---|---|---|---|---|---|---|
| | | Large | Small | No leak | Large | Small | No leak |
| 2,200 | 2.1E−07 | 1.0000 | 0 | 0 | 2.1E−07 | 0 | 0 |
| 2,100 | 3.1E−07 | 1.0000 | 0 | 0 | 3.1E−07 | 0 | 0 |
| 2,000 | 4.5E−07 | 9.999E−01 | 1E−04 | 0 | 4.5E−07 | 4.5E−11 | 0 |
| 1,900 | 6.7E−07 | 9.999E−01 | 1E−04 | 0 | 6.7E−07 | 6.7E−11 | 0 |
| 1,800 | 9.8E−07 | 9.998E−01 | 2E−04 | 0 | 9.8E−07 | 2.0E−10 | 0 |
| 1,700 | 1.4E−06 | 9.993E−01 | 7E−04 | 0 | 1.4E−06 | 1.0E−09 | 0 |
| 1,600 | 2.1E−06 | 9.95E−01 | 5E−03 | 0 | 2.1E−06 | 1.1E−08 | 0 |
| 1,500 | 3.1E−06 | 9.76E−01 | 2.4E−02 | 0 | 3.0E−06 | 7.5E−08 | 0 |
| 1,400 | 4.5E−06 | 8.91E−01 | 1.09E−01 | 0 | 4.0E−06 | 4.9E−07 | 0 |
| 1,300 | 6.7E−06 | 6.90E−01 | 3.10E−01 | 0 | 4.6E−06 | 2.1E−06 | 0 |
| 1,200 | 9.8E−06 | 3.87E−01 | 6.13E−01 | 0 | 3.8E−06 | 6.0E−06 | 0 |
| 1,100 | 1.4E−05 | 1.40E−01 | 8.60E−01 | 0 | 2.0E−06 | 1.2E−05 | 0 |
| 1,000 | 2.1E−05 | 3.1E−02 | 9.69E−01 | 2E−04 | 6.5E−07 | 2.0E−05 | 4.2E−09 |
| 900 | 3.1E−05 | 5E−03 | 9.94E−01 | 1E−03 | 1.6E−07 | 3.1E−05 | 2.5E−08 |
| 800 | 4.5E−05 | 1E−03 | 9.85E−01 | 1.4E−02 | 3.6E−08 | 4.5E−05 | 6.5E−07 |
| 700 | 6.7E−05 | 0 | 8.98E−01 | 1.02E−01 | 8.0E−09 | 6.0E−05 | 6.8E−06 |
| 600 | 9.8E−05 | 0 | 6.13E−01 | 3.87E−01 | 5.9E−09 | 6.0E−05 | 3.8E−05 |
| 500 | 1.4E−04 | 0 | 1.98E−01 | 8.02E−01 | 5.7E−09 | 2.8E−05 | 1.2E−04 |
| 400 | 2.1E−04 | 0 | 1.3E−02 | 9.87E−01 | 4.2E−09 | 2.6E−06 | 2.1E−04 |
| Total | 6.6E−04 | | | | 3.7E−02 | 4.05E−01 | 5.58E−01 |

probabilities for the two cases. The results presented in these tables show that for sequences associated with normal RCS operating pressures (i.e., above 2,000 psig), the system rupture probability is indistinguishable. Indeed, at an RCS pressure of about 1,400 psig, the rupture probabilities are essentially the same. However, at lower RCS pressures, the effect is significant. This is more profound when RCS pressure is below 1,000 psig, where the failure probability can vary by more than two orders of magnitude (from 0.2 to 0.001 at 800 psig). Table 20 compares the effects on the DHR-SD sequence CDF for the two cases. Overall, the sensitivity case produced a CDF for the DHR-SD sequence that is reduced to 33% of its base case value, which would still make it one of the dominant sequences.

**4.6.2 HRA Method Sensitivity.** Because of their relative significance to CDF and risk, two separate sensitivity studies were performed relative to the HRA for this program. The first study relates to the DHR-SD sequence described in the previous section. Instead of examining the effects of pipe rupture pressure uncertainty on CDF, this study identifies the sensitivity of CDF to the assumption on the distribution of HEPs as a function of RCS pressure. That is, how does the probability of the human error of commission (i.e., the premature opening of the DHR letdown

**Table 20.** Sensitivity of pipe rupture pressure uncertainty on DHR-SD sequence CDF (per reactor-year). Base case, log-std-dev = 0.36; sensitivity case, log-std-dev = 0.1.

| Damage state | Base case | Sensitivity case |
|---|---|---|
| OK-op | 3.6E−04 | 3.7E−04 |
| LK-ncd | 3.0E−04 | 2.9E−04 |
| LOCA-ic | 0 | 0 |
| REL-mit | 0 | 0 |
| REL-lg | 1.1E−06 | 3.5E−07 |
| DHR-SD total core damage | 1.1E−06 | 3.5E−07 |

valves, DH-11 and DH-12) vary as a function of RCS pressure. The second HRA sensitivity case studies the effect of optimizing the conditions at the B&W plant with respect to operator performance. In this case, the performance shaping factors are assumed to result in positive influences and produce relatively low HEPs.

***4.6.2.1 Initiation of DHR-SD Sequence.*** The HEP for prematurely opening the DHR letdown valves during shutdown is taken as a cumulative probability that the action is performed before the RCS pressure is reduced to 300 psig. Because the probability of rupturing the DHR system is dependent on the actual RCS pressure, it is necessary to allocate the HEP across the range of possible pressures. In the base case analysis, this HEP is distributed such that the probability of opening the valves at 2,200 psig is about 1,000 times less likely when compared to the value at 400 psig (i.e., relative weights of 0.001 and 1.0, respectively). This sensitivity study identifies the effect of weighing the HEPs more toward the low-pressure end. Specifically, the first sensitivity case uses a relative weighing of 1.0E−04 at 2,200 psig (compared to a weight of 1.0 for 400 psig). Once the relative weighing of the HEPs was estimated, a linear regression line was generated on the log(HEP). This line was then used to calculate the appropriate probability distribution that yielded the estimated cumulative probability of 6.6E−04, as calculated by the HRA. This portion of the process is shown in Tables 21 and 22. The calculation of the pressure dependent HEPs for the sensitivity case is shown in Table 23 (the base case is shown on Table 18). The result of this study is that the CDF is reduced by a factor of approximately two. This sensitivity case represents a likely and realistic alternative to the assumptions made for the base case. Although the base case represents the best estimate for modeling this sequence based on the experience of the analysts and their familiarity with both the subject B&W plant and the historical data, an argument could be made for a less conservative treatment. The results of this sensitivity case provide an estimate of the change in results for the less conservative treatment of this human error, namely a reduction in CDF by a factor of two.

A second sensitivity case was evaluated in which an absolute upper bound on RCS pressure of 1,000 psi was used. In this situation, a linear weighing scheme of between 400 and 1,000 psi provided the basis of the HEP distribution. The regression analysis for this case is presented in Table 24 and the pressure dependent HEPs and rupture probabilities are presented in Table 25. Table 26 lists the results of this analysis along with the base case estimates of CDF and risk for the DHR-SD sequence.

**Table 21.** Relative weighing of HEP as a function of RCS pressure, base case. Regression used for estimating pressure dependent HEPs.

| RCS pressure | Relative HEP | log(HEP) |
|---|---|---|
| 2,200 | 0.001 | -3 |
| 1,600 | 0.01 | -2 |
| 1,000 | 0.1 | -1 |
| 400 | 1 | 0 |

| Regression output | |
|---|---|
| Constant | 0.666667 |
| Standard error of Y estimate | 0 |
| R squared | 1 |
| Number of observations | 4 |
| Degrees of freedom | 2 |
| X coefficient(s) | -0.00167 |
| Standard error of coefficient | 9.6E – 12 |

**Table 22.** Relative weighing of HEP as a function of RCS pressure, sensitivity case. Regression used for estimating pressure dependent HEPs (Sensitivity Case #1).

| RCS pressure | Relative HEP | log(HEP) |
|---|---|---|
| 2,200 | 0.0001 | -4 |
| 1,750 | 0.001 | -3 |
| 1,300 | 0.01 | -2 |
| 850 | 0.1 | -1 |
| 400 | 1 | 0 |

| Regression output | |
|---|---|
| Constant | 0.888889 |
| Standard error of Y estimate | 0 |
| R squared | 1 |
| Number of observations | 5 |
| Degrees of freedom | 3 |
| X coefficient(s) | -0.00222 |
| Standard error of coefficient | 1.2E – 12 |

***4.6.2.2 Optimized PSFs.*** The second HRA sensitivity analysis was conducted to determine if modifications to the human/machine system (performance shaping factors) would result in significant gains in operator performance. This sensitivity case assumes operator performance is improved by modifying procedures (both testing and emergency operating procedures) to explicitly address ISLOCA concerns and through the use of ISLOCA specific training. The performance shaping factors were then re-evaluated assuming plant operations were optimized, and the resulting change in the HEP calculated for the dominant sequences (i.e., DHR-SD, LPI, and MU&P). The following were identified as means for improving the operators' performance in regards to ISLOCA:

- *Procedures* for startup, shutdown, and quarterly stroke test could be modified to include the appropriate cautions, notes, warning, or checklists. An example would be noting the importance of correct valve lineups for HP-27 and HP-29, and the correct lineup for DH-21 and DH-23 in terms of the potential for ISLOCA.

- *Instrumentation* could be improved by including a valve status board in the control room. Also, providing information on pressure, temperature, level, and flow in the interfacing systems in an unambiguous manner to the crew could improve the instrumentation.

**Table 23.** DHR system rupture probabilities (weighted by the HEP of prematurely opening DH-11/12) as a function of RCS pressure. Sensitivity case for relative weight of 1.0E − 04 at RCS pressure of 2,200 psig.

| RCS pressure (psig) | HEP | System rupture probability | | | HEP-weighted system rupture probability | | |
|---|---|---|---|---|---|---|---|
| | | Large | Small | No leak | Large | Small | No leak |
| 2,200 | 2.6E−08 | 1.000 | 0 | 0 | 2.6E−08 | 0 | 0 |
| 2,100 | 4.4E−08 | 9.99E−01 | 1E−03 | 0 | 4.4E−08 | 6.1E−11 | 0 |
| 2,000 | 7.4E−08 | 9.97E−01 | 3E−03 | 0 | 7.3E−08 | 1.9E−10 | 0 |
| 1,900 | 1.2E−07 | 9.95E−01 | 5E−03 | 0 | 1.2E−07 | 5.6E−10 | 0 |
| 1,800 | 2.0E−07 | 9.94E−01 | 6E−03 | 0 | 2.0E−07 | 1.3E−09 | 0 |
| 1,700 | 3.4E−07 | 9.91E−01 | 9E−03 | 0 | 3.4E−07 | 3.0E−09 | 0 |
| 1,600 | 5.7E−07 | 9.83E−01 | 1.7E−02 | 0 | 5.6E−07 | 1.0E−08 | 0 |
| 1,500 | 9.5E−07 | 9.64E−01 | 3.6E−02 | 0 | 9.2E−07 | 3.4E−08 | 0 |
| 1,400 | 1.6E−06 | 9.20E−01 | 8.0E−02 | 0 | 1.5E−06 | 1.3E−07 | 0 |
| 1,300 | 2.6E−06 | 8.36E−01 | 1.64E−01 | 0 | 2.2E−06 | 4.3E−07 | 0 |
| 1,200 | 4.4E−06 | 7.05E−01 | 2.95E−01 | 0 | 3.1E−06 | 1.3E−06 | 0 |
| 1,100 | 7.4E−06 | 5.51E−01 | 4.49E−01 | 0 | 4.1E−06 | 3.3E−06 | 0 |
| 1,000 | 1.2E−05 | 4.03E−01 | 5.97E−01 | 1E−04 | 4.9E−06 | 7.3E−06 | 1.2E−09 |
| 900 | 2.0E−05 | 2.81E−01 | 7.18E−01 | 1E−03 | 5.7E−06 | 1.5E−05 | 1.6E−08 |
| 800 | 3.4E−05 | 1.78E−01 | 8.10E−01 | 1.2E−02 | 6.1E−06 | 2.8E−05 | 4.0E−07 |
| 700 | 5.7E−05 | 1.00E−01 | 8.09E−01 | 9.1E−02 | 5.7E−06 | 4.6E−05 | 5.2E−06 |
| 600 | 9.5E−05 | 5.0E−02 | 5.80E−01 | 3.70E−01 | 4.7E−06 | 5.5E−05 | 3.5E−05 |
| 500 | 1.6E−04 | 2.1E−02 | 1.93E−01 | 7.86E−01 | 3.4E−06 | 3.1E−05 | 1.2E−04 |
| 400 | 2.6E−04 | 7E−03 | 1.2E−02 | 9.81E−01 | 1.7E−06 | 3.3E−06 | 2.6E−04 |
| Total | 6.6E−04 | | | | 6.9E−02 | 2.88E−01 | 6.43E−01 |

**Table 24.** Relative weighing of HEP as a function of RCS pressure, sensitivity case. Regression used for estimating pressure dependent HEPs (Sensitivity Case #2).

| RCS pressure (psi) | Relative HEP | Regression output | |
|---|---|---|---|
| 1,000 | 0 | Constant | 1.66667 |
| 800 | 0.333333 | Standard error of Y estimate | 3.16E−07 |
| 600 | 0.666667 | R squared | 1 |
| 400 | 1 | Number of observations | 4 |
| | | Degrees of freedom | 2 |
| | | X coefficient(s) | -0.00167 |
| | | Standard error of coefficient | 7.07E−10 |

**Table 25.** DHR system rupture probabilities (weighted by the HEP of prematurely opening DH-11/12) as a function of RCS pressure. Assumes a linear probability distribution of the HEP, probability weight at 400 psig = 1.0; at 1,000 psig = 0.0 (Sensitivity Case #2).

| RCS pressure (psig) | HEP | System rupture probability | | | HEP-weighted system rupture probability | | |
|---|---|---|---|---|---|---|---|
| | | Large | Small | No leak | Large | Small | No leak |
| 2,200 | 0 | 1.000 | 0 | 0 | 0 | 0 | 0 |
| 2,100 | 0 | 9.99E−01 | 1E−03 | 0 | 0 | 0 | 0 |
| 2,000 | 0 | 9.97E−01 | 3E−03 | 0 | 0 | 0 | 0 |
| 1,900 | 0 | 9.95E−01 | 5E−03 | 0 | 0 | 0 | 0 |
| 1,800 | 0 | 9.94E−01 | 6E−03 | 0 | 0 | 0 | 0 |
| 1,700 | 0 | 9.91E−01 | 9E−03 | 0 | 0 | 0 | 0 |
| 1,600 | 0 | 9.83E−01 | 1.7E−02 | 0 | 0 | 0 | 0 |
| 1,500 | 0 | 9.64E−01 | 3.6E−02 | 0 | 0 | 0 | 0 |
| 1,400 | 0 | 9.20E−01 | 8.0E−02 | 0 | 0 | 0 | 0 |
| 1,300 | 0 | 8.36E−01 | 1.64E−01 | 0 | 0 | 0 | 0 |
| 1,200 | 0 | 7.05E−01 | 2.95E−01 | 0 | 0 | 0 | 0 |
| 1,100 | 0 | 5.51E−01 | 4.49E−01 | 0 | 0 | 0 | 0 |
| 1,000 | 0 | 4.03E−01 | 5.97E−01 | 1E−04 | 0 | 0 | 0 |
| 900 | 3.1E−05 | 2.81E−01 | 7.18E−01 | 1E−03 | 8.8E−06 | 2.3E−05 | 2.5E−08 |
| 800 | 6.3E−05 | 1.78E−01 | 8.10E−01 | 1.2E−02 | 1.1E−05 | 5.1E−05 | 7.3E−07 |
| 700 | 9.4E−05 | 1.00E−01 | 8.09E−01 | 9.1E−02 | 9.5E−06 | 7.6E−05 | 8.6E−06 |
| 600 | 1.3E−04 | 5.0E−02 | 5.80E−01 | 3.70E−01 | 6.2E−06 | 7.3E−05 | 4.7E−05 |
| 500 | 1.6E−04 | 2.1E−02 | 1.93E−01 | 7.86E−01 | 3.4E−06 | 3.0E−05 | 1.2E−04 |
| 400 | 1.9E−04 | 7E−03 | 1.2E−02 | 9.81E−01 | 1.2E−06 | 2.3E−06 | 1.9E−04 |
| Total | 6.6E−04 | | | | 6.1E−02 | 3.87E−01 | 5.52E−01 |

**Table 26.** Sensitivity of pressure dependent HEP on DHR-SD sequence CDF (per reactor-year). Base case: HEP relative weight at 2,200 psig, 1.0E−03. Sensitivity cases: HEP relative weight at 2,200 psig, 1.0E−04 (Sensitivity Case #1); probability = 0 at 1,000 psig (Sensitivity Case #2).

| Damage state | Base case | Sensitivity Case #1 | Sensitivity Case #2 |
|---|---|---|---|
| OK-op | 3.6E−04 | 4.2E−04 | 3.6E−04 |
| LK-ncd | 3.0E−04 | 2.4E−04 | 3.0E−04 |
| LOCA-ic | 0 | 0 | 0 |
| REL-mit | 0 | 0 | 0 |
| REL-lg | 1.1E−06 | 6.4E−07 | 5.7E−07 |
| Core damage | 1.1E−06 | 6.4E−07 | 5.7E−07 |

- *Training* of the control room and equipment operator personnel could be improved two ways: a formal ISLOCA procedure could be developed and trained upon, and the handling of computerized alarms on the control room could be formalized.

- *Independent checks* (recovery factors) could be included by all tasks covered by procedures having an independent second operator (shift supervisor, instrumentation and control technicians, or maintenance personnel) assigned to signoff on tasks performed.

The HEPs that were calculated based on optimized PSFs were taken from NUREG/CR-1278.[8] Using these sensitivity case HEPs (i.e., optimized), CDF was recalculated and is shown in Table 27. A significant reduction in CDF and risk resulted from the optimization of the HEPs even though the postulated changes to procedures, training, and instrumentation are not believed to be costly.

Table 27. Sensitivity of both CDF (per reactor-year) and risk to optimizing PSFs.

| Risk measure | Base case | Optimum HEPs |
|---|---|---|
| REL-lg | 2.2E−06 | 4.8E−08 |
| REL-mit | 0 | 0 |
| LOCA-ic | 8.9E−08 | 8.0E−09 |
| LK-ncd | 1.5E−03 | 1.2E−03 |
| OK-op | 1.1E−02 | 1.1E−02 |
| Population dose (person-rem, 50-mi) | 6.0 | 0.13 |
| Latent cancers (total grid) | 9.6E−03 | 2.2E−04 |
| Early fatalities | 7.7E−08 | 1.7E−09 |

# 5. OBSERVATIONS

A methodology for evaluating ISLOCA risk has been developed and applied to a reference B&W NPP. This methodology has been successful in providing insights on the *relative* contributions of both hardware faults and human actions to the CDF and risk associated with an ISLOCA. The results indicate that human errors of commission, latent faults of equipment, and normal procedural tasks can combine to produce an ISLOCA sequence. The methodology was also used to identify potential means of reducing these contributions to risk. Lastly, the process has shown that risk significant ISLOCA sequences can occur during nonpower operating modes. The conclusions and recommendations relating to the reference B&W plant are presented below. These conclusions are followed by a preliminary discussion of the relationship of these results to the general population of NPP.

## 5.1 B&W Reference Plant-Specific Observations

The ISLOCA results cannot be compared with those of a previous separate study of the same plant. This is because a PRA for the reference B&W plant is not publicly available. The best results for comparison are from the Brookhaven ISLOCA study for Oconee.[3] The BNL study indicated that the expected CDF from an ISLOCA was about 1E − 06/reactor-year. The B&W reference plant CDF is similar in magnitude. However, the ISLOCA sequences that include only hardware failures (i.e., LPI) were found to be important but not dominant with respect to total plant risk. This is attributed to the exclusion of small leak rate failures (i.e., <200 gpm), and the inclusion of operator actions that would recover the sequence and prevent core damage.

The pressure fragility analysis of the interfacing systems showed that the existing relief valves provide insufficient protection against ISLOCA events. The relief valves in the interfacing systems are designed to mitigate the pressure transients associated with routine valve realignments and pump starts and stops. The pressures generated during the various ISLOCA events cannot be accommodated by the relief capacity of these valves. The RELAP5 calculations (without the relief valves modeled) indicated that equilibrium pressure in the interfacing system was reached virtually instantaneously. The same interfacing systems required less than 10 seconds to reach equilibrium when relief valves were included in the model. A relief valve can be effective in protecting a portion of an interfacing system downstream from a restricting orifice. This is provided that the orifice size was comparable to the relief valve size and the relief valve was also downstream of the orifice.

Based on the pressure fragility and rupture analyses, the DHR heat exchanger was identified as having a relatively high likelihood of being ruptured during an ISLOCA (i.e., a low-pressure fragility). The large diameter, low-pressure pipe was also a very likely candidate for rupture. Specifically, the schedule 10S and 20 pipe on the suction side of the DHR pumps was estimated to have a very high rupture probability in the DHR sequences.

The potential for human errors is influenced by the type of training and procedures used to mitigate the ISLOCA events. Human errors are influenced by the awareness of the potential for ISLOCAs (i.e., the importance of maintaining pressure isolation boundaries during routine plant operations) and the ability to recognize the indications of a breach in the pressure isolation boundary. These two items appear to be the most significant influences on ISLOCA related human errors. At the B&W reference plant of this study, human errors can be influenced by the approach taken at the plant during some operations related to the RCS pressure isolation boundary. These include as follows:

- The quarterly stroke testing of HPI MOV HP-2A results in the tripping of pressure switch PSH-2883A. This pressure switch actuates a computer alarm in the control room. The operation's crew know it is caused by HP-2A test and therefore are unconcerned about its annunciation during testing. This computer alarm is the only direct indication that the RCS is back-leaking into the HPI lines. The implication is that these circumstances may create a situation that increases the potential of the high-pressure computer alarm not being responded to as quickly as it could be.

- In order to deal with the large dead-band on the interlock on valve DH-12, the plant shutdown procedures allow the operations crew to jumper-out the DH-12 interlock if the DH-12 valve will not open when initiating DHR cooling.

The HRA analysis performed to estimate the ISLOCA frequency at the B&W reference plant is influenced by the following:

- The lack of specific procedures for responding to the computer alarm system.

- The training requirements on these beyond design basis severe accident ISLOCA sequences. The information in the plant's training program does not require that the plant staff have a compl te understandin the possibility and consequences of ISLOCA type events.

- Procedures that allow personnel to jumper-out the protective interlock on one of two DHR letdown line MOVs.

- ISLOCA events are not explicitly addressed in the plants emergency operating procedures. However, the alarm response procedures do direct the operators to walk-down accessible portions of the affected systems. Specifically, the procedure for the "LPI low flow" alarm directs the operators to walk-down accessible portions of the LPI/DHR system.

The sensitivity calculations that addressed the effect of optimizing human performance on risk have shown that the estimated ISLOCA risk at the B&W reference plant can be greatly reduced. This risk reduction can be achieved by improving human performance through changes to

- Procedures

- Training

- Indication and alarm response.

A number of positive observations were noted. These o servations are related to the effects associated with interfacing system ruptures and the resulting water spray and accumulation. The ECCS at the B&W reference plant are adequately separated. Each postulated rupture was reviewed based on the premise that all equipment in the same room as the rupture would fail because of the resulting environment. The most compromising situation occurs in the DHP sequences where both trains of the DHR system may fail because of a rupture in the single RCS-letdown line or a rupture in the DHR heat-exchangers (both heat exchangers are in the same room). However, *given that the rupture is isolated*, the plant will be able to rely on the operation of the power conversion system and/or the auxiliary feedwater system to cool the reactor and maintain it in a stable condition. This option was available for every ISLOCA sequence postulated, provided the inter-system rupture was isolated. (Note: In every ISLOCA sequence examined, valves were in place that could be used for this. Thus, the ability to prevent core damage would very likely be available for all ISLOCA sequences).

Conversely, core damage is likely to occur if the interfacing system's rupture is not isolated. This results from small BWST makeup capability (approximately 150 gpm) and the time expected for the ECCS to drain the BWST (approximately 4 to 8 hours).

## 5.2 Generalized Observations

Although the estimated ISLOCA risk and CDF are not of major concern, there is indication that operational ISLOCA-type events are likely. Appendix A lists a number of events that have occurred in the commercial nuclear power industry. In addition, the calculations performed here result in a high likelihood of events being initiated, but a similarly high probability of those events being recovered. The primary reason for the occurrence of these events is believed to be the limited awareness of operations personnel concerning the possibility and consequences of ISLOCA events. Human errors tend to dominate the frequency of the ISLOCA precursors. These human errors that contribute to the ISLOCA risk occur during routine plant operations (e.g., plant startup and shutdown, and periodic surveillance and testing of isolation valves). This human performance results in a relatively high initiating event frequency for ISLOCA sequences. Credit has been given in the analysis to the expertise of the operators of the B&W plant analyzed in this study to being able to recover from a sequence of events that could result in an ISLOCA. It is this consideration that keeps the CDF and risk from being high. The B&W plant used for this study does not have emergency operating procedures that specifically address the ISLOCA scenarios postulated here. The good performance attributed to the control room operators is the result of a "reasonable person" approach. In the HRA, the probability that the operators would successfully recover from a potential ISLOCA scenario was based on how a reasonable person would respond to the indications available in the control room.

(Note: The plant specific B&W emergency operating procedures are not designed to specifically address the ISLOCA sequences. If changes are made to these procedures to address ISLOCA sequences, the ability to recover from these events would be significantly increased. These types of changes should be considered in the context of a severe accident management policy.)

# 6. REFERENCES

1. U. S. Nuclear Regulatory Commission, *Reactor Safety Study—An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400, October 1975.

2. U. S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, Volumes 1 and 2, NUREG-1150, June 1989.

3. G. Bozoli, P. Kohut, and R. Fitzpatrick, *Interfacing Systems LOCA: Pressurized Water Reactors*, NUREG/CR 5102, BNL-NUREG-52135, February 1989.

4. T-L. Chu, S. Stoyanov, and R. Fitzpatrick, *Interfacing Systems LOCA: Boiling Water Reactors*, NUREG/CR 5124, BNL-NUREG-52141, February 1989.

5. J. M. Griesmeyer and L. N. Smith, *A Reference Manual for the Event Progression Analysis Code (EVNTRE)*, NUREG/CR-5174, SAND88-1607, September 1989.

6. D. A. Wesley et al., *Pressure-Dependent Fragilities for Piping Components*, NUREG/CR-5603, EGG-2607, October 1990.

7. T. G. Ryan, "A Task Analysis Linked Approach (TALENT) for Integrating the Human Factor in Reliability Assessment of NPP," *Reliability Engineering and System Safety, 22*, 1988.

8. A. D. Swain and H. E. Guttmann, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications*, NUREG/CR-1278, SAND80-0200, August 1983.

9. G. Hannaman and A. Spurgin, *Systematic Human Action Reliability Procedure (SHARP)*, EPRI NP-3583, Electric Power Research Institute, 1984.

10. Institute of Electrical and Electronics Engineers, *A Guide for General Principles of Human Action Reliability Analysis for Nuclear Power Generation Stations*, IEEE Draft Standard P1082/D7, August 1989.

11. D. I. Gertman et al., *Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)*, NUREG/CR-4639, December 1987.

12. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, *ISLOCA Inspection Report*, Report No: 50-346/89-201, inspection conducted Oct. 30–Nov. 9, 1989.

13. R. C. Bertucio et al., *Analysis of Core Damage Frequency: Sequoyah, Unit 1, Internal Events*, NUREG/CR-4550, SAND86-2084, Rev.1, Vol. 5, June 1989.

14. K. D. Russell et al., *Integrated Reliability and Risk Analysis System (IRRAS), Version 2.5*, NUREG/CR-5300, EGG-2613, Vol. 1, February 1991.

15. Science Applications International Corporation, *ETA-II User's Manual*, Los Altos, California, November 4, 1987.

16. D. C. Aldrich et al., *Technical Guidance for Siting Criteria Development*, NUREG/CR-2239, SAND81-1549, 1982.

17. Nuclear Safety Analysis Center (Electric Power Research Institute) and Duke Power Company, *Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3*, NSAC-60, June 1984.

18. F. J. Rahn, J. Collen, and A. L. Wright, "Aerosol Behavior Experiments on Light Water Reactor Primary Systems," *Nuclear Technology, 81*, May 1988, pp. 158–182.

19. D. L. Hagrman, *A Study of Fission Product Transport from Failed Fuel During N Reactor Postulated Accidents*, EGG-NERD-8541, September 1989.

20. D. I. Chanin et al., *MELCOR Accident Consequence Code System (MACCS)*, NUREG/CR-4691, February 1990.

21. K. E. Carlson et al., *RELAP5/MOD3 Code Manual Volume I: Code Structure, System Models, and Solution Methods*, NUREG/CR-5535, EGG-2596, Draft, (available from EG&G Idaho, Inc., P.O. Box 1625, Idaho Falls, ID 83415-2404) June 1990.

22. K. K. Murata et al., *User's Manual for CONTAIN 1.1: A Computer Code for Severe Nuclear Reactor Accident Containment Analysis*, NUREG-CR-5026, SAND87-2309, November 1989.

23. R. Steele, Jr., K. G. DeWall, and J. C. Watkins, *Generic Issue 87: Flexible Wedge Gate Valve Test Program*, NUREG/CR-5558, EGG-2600, January 1991.

24. R. P. Kennedy et al., "Probabilistic Seismic Safety of an Existing Nuclear Power Plant," *Nuclear Engineering and Design, 59*, No. 2, August 1980.

25. J. Reason, *Human Error*, Cambridge University Press: New York, NY, 1990.

26. A. S. Benjamin et al., *Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1*, NUREG/CR-4551, SAND86-1309, Vol. 1, (draft report for comment), February 1987, (available from Public Document Room, Washington D.C.).

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11. ABSTRACT (200 words or less)

This document presents information essential to understanding the risk associated with inter-system loss-of-coolant accidents (ISLOCAs). The methodology developed and presented in this document provides a state-of-the-art method for identifying and evaluating plant-specific hardware designs, human performance issues, and accident consequence factors relevant to the prediction of the ISLOCA risk. This ISLOCA methodology was developed and then applied to a Babcock and Wilcox (B&W) nuclear power plant. The results from this application are described in detail. For this particular B&W reference plant, the assessment indicated that the probability of a severe ISLOCA is approximately 2.2E – 06/reactor-year.

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